Nuclear Safety
NEA/CSNI/R(2015)12
November 2015
www.oecd-nea.org

International Workshop on Fire Probabilistic Risk Assessment (PRA)

Workshop Proceedings
Garching, Germany
28-30 April 2014

OECD
BETTER POLICIES FOR BETTER LIVES

NEA
NUCLEAR ENERGY AGENCY

Unclassified **NEA/CSNI/R(2015)12**

Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development

English text only

NEA/CSNI/R(2015)12
Unclassified

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

International Workshop on FIRE Probabilistic Risk Assessment (PRA)

Workshop Proceedings

Garching, Germany
April 28 - 30, 2014

English text only

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation’s statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

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

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was created in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety among NEA member countries. The main tasks of the CSNI are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and reach consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The priority of the committee is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs, the committee provides a forum for improving safety-related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operative mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA), which is responsible for the Agency's programme concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the other NEA Standing Technical Committees as well as with key international organisations such as the International Atomic Energy Agency (IAEA) on matters of common interest.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

On April 28-30, 2014, CSNI Working Group on Risk Assessment (WGRISK) held a workshop on Fire Probabilistic Risk Assessment (PRA) in Garching, Germany. The meeting was attended by nearly 60 participants from 15 countries, the NEA and the International Atomic Agency (IAEA). The participants' organisations included licensees, regulators, and technical support organisations. A majority of the attendees were experienced in the purpose and performance of Fire PRA. Some were involved in related activities (e.g., deterministic fire modelling, fire experiments).

The workshop objectives were:

- to support assessment of current state of probabilistic analyses of fire hazards for nuclear installations at the design stage and during all plant operational states from start of operation up to the longer lasting post-commercial operating phases,
- to support re-evaluation of Fire PRA, in particular as a tool to address the lessons learned from the post-Fukushima investigations and stress tests with respect to fire events,
- to share methods as well as good practices and experiences among member states on probabilistic risk assessment of fire hazards and event combinations with fires,
- to identify new potential topics for further WGRISK activities in this area, including potential update of the State-of-the-art Report (SOAR) on fire risk analysis.

Following the workshop, a writing session involving the session chairs and members of the task core group was held with the goal to prepare an initial draft of this report, which would summarise not only the workshop contents and presentations but also important ideas raised during the workshop. It was also intended to provide a list of conclusions and recommendations for identifying possible further actions of WGRISK and other working groups and projects with regard to topics related to the fire risk in nuclear installations.

There are a wide number of detailed conclusions set out in Section 4 of the report according to the workshop's objectives. Importantly, the main conclusion of this work is that based on the workshop presentations, discussions during the sessions (including the opening and final sessions), and the post-workshop discussion and writing session, it appears that Fire PRA has achieved a reasonable level of maturity. Internationally, Fire PRAs are performed within a consistent framework and many employ common methods, tools, and guidance. There appears to be a common understanding of weaknesses and work is ongoing to support key areas. Most importantly, despite the recognised weaknesses, the results of Fire PRA are more or less being used to support risk-informed decision making by plant owners and operators as well as by regulators.

Based on its review of the workshop conclusions and subsequent discussions, WGRISK has the following recommendations.

- Recognising that fire-related risk-informed applications are likely to remain important over the next few years, that the scope of such applications is increasing (e.g., to include Level 2 PSA and low power and shutdown operations), and that Fire PRA research and applications are resulting in technological improvements, WGRISK should continue to remain active in the Fire PRA area. Potential (and non-exclusive) future activities include:

  - the organisation of a future workshop, perhaps focused on a specific aspect of Fire PRA, including:
    - multiple hazards/events (including fire),
    - fire human reliability analysis (HRA),
    - multi-unit Fire PRA,
  - development of additional resource or guidance documents (e.g., a SOAR, a condensed SOAR, or a Technical Opinion Paper (TOP), a focused guidance document/database based on member country experiences), as discussed in the main body of this report;
  - participation (perhaps as a peer review organisation or as a source of peer reviewers) in international comparison and harmonisation activities (e.g., the European Union's ASAMPSA_E Project); and
  - participation in future conferences and other professional meetings.

Specific future activities will be pursued by WGRISK following its normal processes (i.e., developing a CSNI Activities Proposal Sheet (CAPS), identifying a core group of participants, gaining working group approval, and gaining CSNI approval).

- Recognising the value of risk information in focusing operational experience data collection and analysis, and the value of operational experience data in shaping the development of PRA models and in the quantification of PRA model parameters, WGRISK should:
  - strengthen its ties with international operational experience working groups particularly the NEA Committee of Nuclear Regulatory Authorities Working Group on Operating Experience (WGOE); and
  - support efforts to more broadly disseminate risk-relevant lessons from operating experience reviews (e.g., such as those discussed at the annual Technical Meeting on Experiences with Risk-based Precursor Analysis held in Belgium, whose conduct and findings may not be widely known within the PSA community).

(Although the latter recommendation is not specific to Fire PRA, it is the outcome of general discussions held during the workshop and naturally incorporates it).

- As indicated in the main body of this report, WGRISK has carried out numerous activities in Fire PRA. However, it has not tracked the outcomes and recommendations (detailed as well as general) resulting from these activities. Therefore it is recommended that WGRISK should develop and implement a process to systematically record and track activity conclusions and recommendations, in order to ensure the full knowledge of past efforts is built into future activities.

- Finally, although the role of WGRISK is to share information and experience related to risk assessment it is recognised that this work can also be potentially useful direct input to nuclear safety and the development of safety standards used by member organisations. Thus, WGRISK should strengthen its links with standards development organisations including IAEA (which participates in WGRISK meetings). This would likely produce mutual benefit to WGRISK (both in ensuring use of its products and in identifying areas of need) and these other organisations (e.g. by providing information supporting improved standards and guidance).

1. INTRODUCTION

1.1 Background

Deterministic fire hazard analyses (FHA), investigations of the operating experience at nuclear installations and Fire Probabilistic Risk Assessment (PRA[1]) have demonstrated that knowledge of the frequency of the incidence of fires is an important contributor to nuclear power plant (NPP) fire risk assessment. However, fires and associated plant responses are complex phenomena, and so the estimates of fire risk are subject to considerable uncertainty. In an attempt to improve the situation the CSNI (Committee on the Safety of Nuclear Installations) Working Group Risk (WGRISK, formerly PWG5) established a task group in 1996 to review the status and maturity of methods used in fire risk assessment for operating NPPs. These activities resulted in an international workshop carried out in 1999 in Helsinki, Finland (cf. [1]). Workshop participants exchanged information regarding risk related insights from fire analyses, existing methods, and available data for Fire PRA. In 2000, the NEA published the task group's State-of-the-art Report (SOAR) on "Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics," [2], whose conclusions included the statement "*The shortage of fire analysis data is one of the major deficiencies in the present fire risk assessment*".

Based on the SOAR conclusions, several OECD member countries agreed to establish several activities in this field, such as the international Fire Incidents Records Exchange (FIRE) Project [3] as well as the experimental Project PRISME (French acronym for propagation d'un incendie pour des scénarios multilocaux *é*lémentaires) [4] under the umbrella of the CSNI to encourage multilateral co-operation in the collection and analysis of data related to fire events in NPPs and investigating nuclear plant related fire scenarios relevant to be simulated within fire safety and risk analyses. Summarised information on these projects and related topics can be found in Section 2.

Since the publication of NEA/CSNI/R(99)27 [2], Fire PRA methods, tools, and data have been significantly enhanced, and Fire PRA has become a mandatory tool for supporting safety reviews of NPPs in several member countries, see [5], [6] and [7]. Fire PRA is required by the IAEA, e.g., as discussed in SSG-3 [8], and by the Reactor Safety Reference Levels of the Western European Nuclear Regulators' Association (WENRA) Reactor Harmonisation Working Group (RHWG) [9].

The follow-up WGRISK Fire PSA workshop in Mexico in 2005 [10] indicated that important new developments had taken place and also addressed needs for further development. Examples of developments since that workshop include the following standards and guidelines, in particular from the U.S. and Germany:

- "*Fire PRA Methodology for Nuclear Power Facilities*" (EPRI 1011989 and NUREG/CR-6850,[2] Volumes 1 [11], 2 [12] and Supplements [13]),

- "Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME/ANS RA-Sa-2009) [14] superseding the former ANSI/ANS-58.23-2007 "Fire PRA Methodology",

---

1 In this report the abbreviations PRA (Probabilistic Risk Assessment) and PSA (Probabilistic Safety Assessment) are used synonymously.

2 This report was developed jointly by the Electric Power Research Institute (EPRI) and the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research. For brevity, the report will be referred to as "NUREG/CR-6850" in the remainder of this workshop summary.

- "*Verification and Validation of Selected Fire Models for NPP Applications*" (EPRI 1011999 / NUREG-1824) [15],

- "Nuclear Power Plant Fire Modelling Analysis Guidelines (NPP FIRE MAG) — Final Report" (EPRI 1023259 / NUREG-1934) [16],

- *"A Framework for Low Power/Shutdown Fire PRA – Final Report*" (NUREG/CR-7114) [17],

- "*EPRI/NRC-RES Fire Human Reliability Analysis Guidelines*" (EPRI 1019196 / NUREG/CR-1921) [18],

- the German PSA Guide with its supplementary technical documents on *PSA Methods* (BfS-SCHR-37/05) [19] and *Data* (BfS-SCHR-38/052) [20], and the corresponding *"Addenda on PSA Methods and Data for Nuclear Power Plants"* [21] on enhancements in PSA methodology covering in particular Fire PRA for low power and shutdown plant operational phases.

Numerous experimental programmes have been performed to support these standards and guidance as well as future Fire PRA methods developments. Recently completed activities include: CAROLFIRE [22], CHRISTIFIRE [23], KATE-Fire [16], JACQUE-FIRE [25], [26] in the U.S. or the international experimental project PRISME [4], [28]. Ongoing experimental activities include the OECD High Energy Arcing Fault (HEAF) programme [29], [30] and the PRISME2 program [31]. Work has also been done to enhance fire simulation codes, such as the French zone code SYLVIA (simulation of consequences of a fire in an industrial facility featuring a ventilation network) and the French CFD (*c*omputational *f*luid *d*ynamics) type code for fire simulation ISIS, developed based on an experimental fire research programmes, the German lumped parameter type code COCOSYS, and the open source CFD type fire simulation code FDS (*F*ire *D*ynamics *S*imulator). Code verification and validation has been addressed by several national as well as international projects, as documented in [15], [4], [29], [32]. As another example, in Finland joint probabilistic-deterministic fire spreading scenarios have been developed using the FDS code, specifically for application to I&C cabinet rooms, cable tunnels and cable spreading rooms [**35**]. Ongoing efforts to collect and analyse operational fire events include the EPRI Fire Events Database project [**34**] and the previously mentioned OECD FIRE Database project [**3**]. Additional information on these activities can be found in Section 2.

Moreover, WGRISK activities in the recent past have revealed the impression that several countries are heading towards a ”Living PSA” including both Level 1 and Level 2, for full power as well as low power and shutdown plant operational states, covering both internal and external hazards and events, which includes fires as internal hazards. Recognising the value of sharing information on these and other international developments in light of the recent high-visibility applications of Fire PRA (e.g., in support of U.S. transitions from deterministic to risk-informed, performance-based fire protection programmes), WGRISK developed a proposal to hold a Fire PRA workshop (see Appendix 1). This proposal was accepted by CSNI in June, 2012.

A task group for this international Fire PRA workshop was formed under the leadership of the WGRISK Chair from GRS, Germany and STUK, Finland having carried out the first workshop in 1999. The task group with additional members from Finland, France, United States of America and the NEA developed a workshop programme addressing CSNI challenges and associated technical goals such as further review and assessment of developments in PSA methodology and approaches related to radioactivity confinement, criticality, fire and chemical risks in nuclear installations as well as contributions to the enhancement of safety performance of currently operating nuclear installations.

The workshop, which was originally planned for 2013, was held in 2014 to ensure strong participation by active member countries.

1.2 Objectives of the workshop

The specific workshop objectives are provided in the workshop announcement (see Appendix 2):

- to support assessment of current state of probabilistic analyses of fire hazards for nuclear installations at the design stage and during all plant operational states from start of operation up to the longer lasting post-commercial operating phases,
- to support re-evaluation of Fire PRA, in particular as a tool to address the lessons learned from the post-Fukushima investigations and stress tests with respect to fire events,
- to share methods and good practices and experiences among member states on probabilistic risk assessment of fire hazards and event combinations with fires,
- to identify new potential topics for further WGRISK activities in this area, including potential update of the State-of-the-art Report (SOAR) on fire risk analysis.

1.3 Organisation of the workshop

The workshop, which was held April 28-30, 2014, was organised by a core group of WGRISK members from Germany (the task lead), Finland, France and the U.S. The workshop scope, as indicated in the announcement, covered a broad range of topics:

- Progress in Fire PRA methodology (overview on relevant issues, methods and data):
  - Screening approaches in the frame of Fire PRA;
  - Fire PRA methodology for low power and shutdown states (including long-term post-commercial operation phases, planned as well as unintended shutdown states);
  - Potential of combinations of fire events with other events and/or hazards and their treatment and consideration in the analysis;
  - Treatment of epistemic (knowledge related) as well as aleatory (stochastic) uncertainties in fire risk assessment;
  - Improvements/Enhancements with respect to data for fire risk analysis;
  - Fire experiments;
- Modelling of fire event sequences for PRA use**:**
  - General enhancements in fire modelling for nuclear facilities;
  - Modelling of complex multi-room fire scenarios under different ventilation conditions, including probabilistic treatment of fire spreading;
  - Fire suppression modelling and modelling of manual firefighting activities;

  - Simulation of fire events as dynamic processes (in time and physical characteristics), particularly for complex geometries and varying boundary conditions;

  - Treatment of uncertainties in fire simulations;

  - Sensitivity and importance analyses for Fire PRA;

- Impact of direct, indirect and consequential fire effects, in particular on safety-related systems and equipment:

  - Analysis of local and global effects of fire hazards including long-term loss of the electrical grid and the ultimate heat sink;

  - Treatment of fire induced failures such as electrical faults;

  - Evaluation of the effectiveness of measures to be taken in response to induced fire hazards (e.g., seismically-induced fires);

  - Human factors and organisational factors related to fire risk assessment, including effects of multiple units and events combinations;

- Use of Fire PRA in risk-informed decision making:

  - Applications of Fire PRA approaches in recent regulations;

  - Use of Fire PRA results in risk-informed inspection and maintenance (by the licensees, including other non-regulatory applications of Fire PRA);

  - Plant safety modifications and improvements using Fire PRA results;

  - Defence-in-depth considerations for risk-informed fire protection approaches;

- Extension of Fire PRA scope and applications:

  - Consideration of event combinations of fires and other internal and external hazards (seismic, flooding, aircraft crash, HEAF, explosions, etc.);

  - Extension of Fire PRA for Level 2 PRA analyses;

  - Fire PRA applicability and extension of applications.

The actual programme (see Appendix 2), which reflected the papers submitted by the participants, included an opening session, eight technical sessions devoted to participant presentations and short facilitated discussions, and a concluding session with summaries of the technical presentations (delivered by the session chairs) and a general discussion.

The meeting was attended by nearly 60 participants from 15 countries, the NEA and the IAEA. (The meeting attendance list is included as Appendix 3.) The participants' organisations included licensees, regulators, and technical support organisations. Most of the attendees were experienced in the purpose and performance of Fire PRA. Some were involved in related activities (e.g., deterministic fire modelling, fire experiments). The workshop also had a number of observers from the host organisation (GRS).

Following the workshop, a one-day writing session involving the session chairs and members of the task core group was held with the goal to prepare an initial draft of this report, which would summarise not only the workshop contents and presentations but also important ideas raised during the workshop. It was also intended to provide a list of conclusions and recommendations for identifying possible further actions of WGRISK, and other working groups and projects with regard to fire and PSA related topics, such as WGIAGE, WGAMA, WGHOF, WGOE, PRISME2 and OECD FIRE.

2. ACTIVITIES PRECEDING ORGANISATION OF THE WORKSHOP

In the context of fire probabilistic risk analysis and assessment WGRISK has sponsored or co-sponsored a number of PSA related activities over many years prior to this workshop. These activities also included several projects and workshops, which were completely or partially devoted to fire risk:

- an international workshop on Fire PSA in 1999 [1];
- the creation of a Fire Incidents Records Database [3];
- a follow-up Fire PSA workshop in 2005 [9];
- the PRISME experimental project from 2006 to June 2012 [4];
- the ongoing PRISME2 experimental project started in June 2012 [31];
- the WGIAGE task on High Energy Arcing Fault Events (HEAF) [29];
- the High Energy Arcing Fault (HEAF) Events experimental project [**30**];
- the common WGRISK and Database Working Groups´ task on the *"Use of OECD Data Project Products in Probabilistic Safety Assessment"* [35].

The results of information exchange in the frame of the WGRISK are compiled in a CSNI report entitled "The Use and Development of Probabilistic Safety Assessment" first issued in 2002 [5], updated in 2007 [6] and 2012 [7]. Fire PSA is one of the topics covered by the report. The most recent version notes with respect to Fire PSA that many countries have already extended or are extending Level 1 PSA to address internal fires and that there are efforts underway to widen the scope of Fire PSA methodology.

The NEA joint database projects and information exchange programmes enable interested countries to pursue research or the sharing of data with respect to particular areas or problems. The following database projects have direct relevance to PSA activities:

- International Common Cause Failure Data Exchange (ICDE) which also subsumes the former NEA Computer-based System Important to Safety (COMPSIS) Project,
- NEA Fire Incidents Record Exchange (FIRE) Project, and
- Component Operational Experience, Degradation and Ageing Programme (CODAP) which subsumes the former OECD Piping Failure Data Exchange (OPDE).

These projects in principle support the collection and analysis of data relevant in the development and review of PSA models, particularly in the areas of material degradation and ageing, common cause failures, fire risk, and digital instrumentation and control systems. Moreover, several of these projects include specific objectives to support quantification activities. However, to date, WGRISK members, particularly those who are not members of the Database Projects, have made little use of the data project products (principally reports). To address this challenge, and based on needs expressed by a number of member countries, WGRISK carried out a task on "*Use of OECD Data Project Products in Probabilistic*

Safety Assessment" in NEA member countries coordinated with representatives from all database projects including the one on fire event data (FIRE) and benefitted greatly from the perspectives offered by these.

The final task report [35] provides the survey responses and associated analysis, along with a detailed description of the key attributes of each of the data projects. The report also includes recommendations for strengthening collaboration between the PSA community and the joint data projects such as FIRE. Best practices for the use of data project products, e.g. fire events data, for PSA, have been identified, along with a summary of success factors for data project activities.

In general, the NEA joint Database Projects represent mature data collection efforts and have enjoyed substantial support from the NEA membership. These projects have endeavoured to ensure that data collection activities have a high level of completeness and quality. This commitment to quality has resulted in the development of project-specific programmatic requirements intended to ensure quality. However, there remain some challenges when attempting to apply data project products to PSA activities (e.g., data completeness and exposure information needed to calculate PSA parameters). As such, data applicability and completeness should be fully assessed prior to applying data project products to a specific application. Despite these challenges, experience has been developed by a number of NEA members in applying e.g., data from the FIRE Database to PSA initiatives. Examples include fire frequency calculation, and estimation of piping rupture frequencies. Overall, the data projects are an important NEA activity, particularly for member states with a small number of nuclear installations and limited national databases.

To provide additional context for the workshop, brief summaries of a number of activities relevant to Fire PRA are provided below. These include:

- NEA activities preceding the organisation of this workshop which had already provided valuable insights of the necessity for such an activity,
- recently completed and ongoing fire risk related projects by NEA and
- other, mainly experimental, projects linked to the context of the workshop objectives.

2.1 NEA fire risk related experimental projects

PRISME and PRISME2 experimental projects

The OECD PRISME Project ("Propagation d'un incendie pour des scénarios multilocaux élémentaires") [4], [27] and [28] was an experimental programme investigating the spread of a fire and fire effects in multi-room scenarios. The experimental programme was carried out by IRSN, France, between 2006 and 2012. There were partners from twelve member countries.

The objective of the PRISME Project was to study the mechanisms affecting the propagation of hot gases and smoke from a burning room to adjacent rooms through various pathways (open doors, leakages, firebreak doors, air vents, etc.) under the action of mechanical ventilation. A second objective was to investigate the performance of active components (fire suppression and fire dampers) in integrated multi-room scenarios with complex fire sources. The program consisted of four test campaigns:

1. SOURCE to characterise the fire source in open and single-room environments. The main results from this campaign were related to the effects of the vitiated environment and ventilation rate on the burning rate of liquid pool fire. The results also served as a reference to the remaining tests.

2. DOOR to quantify the heat and smoke transport through open doors in multi-room fire scenarios with mechanical ventilation.

3. LEAK to quantify the heat and smoke transport through mechanical leakages, defective fire door and mechanical ventilation system.

4. INTEGRAL to quantify the effects of the geometrical complexity (two, three or four rooms), burning material (liquid fuel, electrical cables or electrical cabinets), and active fire protection measures (water based suppression and fire dampers) on the fire induced conditions in closed and mechanically ventilated compartments.

The PRISME Project increased the understanding of the liquid pool burning behaviour under vitiated conditions and the relative importance of different fire effects spreading mechanisms. The project produced high quality data for the purpose numerical simulation code validation, to be used by the project partners and OECD member countries.

In parallel to the experimental project, an analytical working group was organised to discuss the code validation issues. The participation of the partners was on voluntary basis.

The objective of the ongoing OECD PRISME2 Project is to investigate the topics that did not have sufficient coverage in the first PRISME Project. IRSN remains as operating agent, most of the partner countries are also the same. PRISME2 started in 2012 for a five years period.

PRISME 2 consists of four experimental campaigns

1. Vertical Smoke Propagation (VSP) to investigate the spreading of heat and smoke through vertically stacked rooms, and experimental scenario rarely investigated outside the nuclear community;

2. Cable Fire Spreading (CFS) to characterise the burning behaviour of horizontal cable trays in open and mechanically ventilated atmospheres. Two different types of cable material are included the experiments: halogenated polymer (PVC) and non-halogenated flame retardant cables;

3. Fire Extinction Systems (FES) to assess the performance of water-based suppression systems under different fire conditions;

4. Provisional fire tests to investigate individual topics appearing during the project, such as the efficiency of fire stopping materials on cable trays, effect of tray inclination, and an initial additional test to determine the ventilation effects on PVC cables.

HEAF experimental programme

Another experimental project with high significance for estimating the fire-related risk is the High Energy Arcing Fault (HEAF) Events Project [30]. The objective of this ongoing experimental programme, following an NEA/CSNI/WGIAGE task collecting insights on this type of events, their basic phenomena and the effects on the plants´ safety [29], is to obtain scientific fire-related data on the HEAF phenomenon known to occur in NPPs through carefully designed experiments. The programme period is intended to be July 2012 through December 2015.The details of the experimental programme were established in early 2014. Additional information is provided in a paper presented at the workshop by N. Melly (NRC, USA), provided in Appendix 4.

2.2 NEA fire risk related non-experimental projects

OECD FIRE database Project

The OECD FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) Database is one of the NEA databases collecting events from the operation of NPPs. In the late 1990s it became evident that the only international database collecting fire events from nuclear installations, the IRS (*I*nternational *R*eporting *S*ystem) provided by the IAEA, was not collecting information suitable for specific analysis and use in risk assessment, as only events impacting items important to safety or human health are reported to it. It was recognised that in line with the use of dedicated databases to collect lessons learned on specific topics and quantitative analysis, there was a significant need for a fire events database to inform Fire PRA.

The purpose of the FIRE Database Project is to provide a platform for multiple countries to collaborate and exchange fire events data and thereby to enhance the knowledge of fire phenomena on the one hand and, on the other hand, to improve the quality of risk assessments requiring fire-related data and knowledge. Applicable to commercially operated NPPs only, the OECD FIRE Database covers a collection of data from fire events from all plant operational modes, also including construction, post-commercial operation shutdown and decommissioning phases.

The FIRE Project was formally launched in 2003 for a three-year period with nine participating member countries, which was followed by two four years terms with the addition of three further countries. Phase Four has already started in 2014, and members anticipate a further term for the Project to be started in 2016.

The objectives of the OECD FIRE Project include the establishment of a framework for multi-national co-operation in sharing event information useful to fire risk assessment. The primary activity was to define the format for collecting fire event experience in a quality assured and consistent database. In the course of the Project improvement of fire record event attributes was made to facilitate quantification of fire frequencies and fire risk analysis. The core activity is to collect and analyse fire events over the long term so as to better understand such events, their causes, and their prevention. The Database thus obtained allows the generation of qualitative insights into the causes of fire events which can then be used to derive approaches or mechanisms for their prevention or for mitigating their consequences. Among the applications of the Database is the possibility for member countries to establish a mechanism for the efficient feedback of experience gained in connection with fire events, including the development of defences against their occurrence, such as improvements of the existing national as well as international reporting systems and indicators for risk-based inspections.

The FIRE Database represents a valuable tool for facilitating the use of fire experience of NPPs in the member countries for FIRE PSA. The current version of the Database [36] contains 438 fire events, the wide majority of them quality assured. The events are from the period from the early 1980s to the end of 2013, with the bulk of the events in the period of the mid-1990s to the end of 2013. Although the reporting of events is not exhaustive, the Database provides a good platform for first use within Fire PSA as well as for deterministic investigations.

It is possible to quickly estimate, according to the analytical task to be performed, fire frequencies for different samples of fire events for all plant operational states, different types of reactors, selected sets of countries under consideration of reporting criteria and thresholds in member countries. A variety of applications have already started to make increased use of newly added enhancements for interactive queries and evaluation tasks and of the enhanced statistical possibilities.

Data collection and use for fire risk assessment applications is continuing.

2.3 Other fire risk related experimental projects

In the following, a brief overview is given on recent past or ongoing fire risk related experimental projects and programmes outside the international activities of the NEA.

CAROLFIRE

The Project CAROLFIRE (Cable Response to Live Fire) [22] by US NRC included a series of 78 small-scale tests, and a second series of 18 intermediate-scale **o**pen burn tests. The tests were designed to complement previous testing and to address two needs; namely, to provide data supporting resolution of regulatory issues associated with risk-informed approaches for Post-Fire Safe-Shutdown Circuit Inspections, and improvements to fire modelling in the area of cable response to fires

CHRISTIFIRE

The overall goal of the US NRC CHRISTIFIRE (Cable Heat Release, Ignition, and Spread in Tray Installations during Fire) programme [23] was to better understand and quantify the burning characteristics of grouped electrical cables commonly found in nuclear power plants. The first phase of the programme focused on horizontal tray configurations. The experiments conducted ranged from micro-scale, in which very small (5 mg) samples of cable materials were burned in a calorimeter to determine their heat of combustion and other properties; to full-scale, in which horizontal arrays of ladder-back trays loaded with varying amounts of cable were burned under a large oxygen-depletion calorimeter.

KATE-Fire

The U.S. NRC project “Kerite Analysis in Thermal Environment of Fire” (KATE-Fire) [24]. This NRC project included cable functionality tests conducted on the cable product marketed under the trade name Kerite FR. Although Kerite FR is a thermoset polymer, reviews have identified prior tests that documented cable failure at relatively low temperatures compared to other known thermoset insulation materials. The project report summarises the previously available information for this cable product, presents all of the newly developed test data and provides an assessment of the material’s fire exposure electrical performance.

JACQUE-FIRE

The U.S. NRC JACQUE-FIRE (Joint Assessment of Cable Damage and Quantification of Effects from Fire) program [25], [26] involved eliciting expert judgements from a PRA panel, building upon information developed by an electrical engineering expert panel in a Phenomena Identification and Ranking Table (PIRT) exercise. The objective of the PRA panel’s expert elicitation was to estimate the conditional probabilities of the hot short-induced spurious operation failure mode of different control circuits and their durations, given fire induced cable damage. The results of this programme are intended for use in Fire PRA applications. Distributions were developed for control circuit configurations where applicable test data are available, as well as configurations for which test data are not available.

3. SUMMARY OF THE WORKSHOP ON FIRE PRA

The workshop included an opening session, eight sessions with participant presentations followed by short discussions, and a concluding session with a facilitated discussion. The contributions were devoted to new methodological developments, research activities, use and application of fire event databases, projects with Fire PRA, and requirements for Fire PRA.

3.1 Opening session

The opening session started with a welcome address by Andreas Schaffrath, Division Head of GRS in Garching. The welcome address was followed by the introduction and opening remarks by Marina Röwekamp from GRS as task group leader and organiser of the workshop, a keynote speech by Nathan Siu (NRC, USA) as former WGRISK Chair under whose chairmanship the task was started and a presentation by the NEA Technical Secretariat, Neil Blundell.

The keynote speech titled "*Fire PRA: A Brief History and Some Workshop Challenges*" covered a brief overview of five "near miss" fire events, the U.S. Fire PRA history starting in 1975 with the Browns Ferry fire and WASH-1400, and the current activities with NUREG/CR-6850 [11], [12], [13] and NFPA 805 [37] licence application requirements (LARs). Furthermore, the fire risk was put in perspective to other risk contributors showing that fire risk could be a significant risk driver. Reference was given to past and ongoing NEA efforts in the area including the previous Fire PSA workshops in Helsinki (1999) and Mexico (2005). The presentation reminded the participants that the intent of the workshop was to:

- support assessment of current state of Fire PRA;
- share methods and good practices and experiences;
- identify new potential topics for further WGRISK activities.

and challenged the participants to participate actively, thereby helping WGRISK separate "signal" from "noise."

The presentation by Neil Blundell was entitled "*The Structures and Processes that Support Projects and Working Groups to Deliver Good Practice Guidance*". The presentation covered the structure of the NEA and its standing committees, the roles of the standing committees with safety mandates, the CSNI Joint Safety Research Projects and the relationship with the NEA databases, Probabilistic Safety Assessment, WGRISK and cross cutting issues as well as relevant structure of the NEA Secretariat, its role and how to access it. Finally, the importance of sharing information was stressed.

3.2 Session 1 – Progress in Fire PRA methodology – Relevant issues and methods

Three papers were presented in Session 1 on the recent progress concerning Fire PRA methodology. The first paper provided a general discussion on the maturity and realism in Fire PRA and presented several questions related to realism and maturity to be thought of during the workshop and for the future. The second presentation talked about a comparison of room-based versus component-based approaches for estimating fire frequencies. Finally, the third paper discussed temperature-based filter (or screening) criteria with respect to temperature impact to for use in Fire PRA.

The first paper was presented by Nathan Siu (NRC, USA) on the topic "*Fire PRA Maturity and Realism: A Technical Evaluation and Questions*". Reference was given to the three generations of Fire

PRA from early industry studies (Zion, Indian Point, etc.) and NUREG-1150 [38], through the IPEEEs (*I*ndividual *P*lant *E*xaminations of *E*xternal *E*vents) requested by the NRC in Supplement 4 to Generic Letter 88-20 issued in 1991, to current risk-informed, performance-based fire protection activities (following NFPA 805 [37] and supported by NUREG/CR-6850 [5], [12], [13] and related Fire PRA guidance). The paper noted concerns voiced in the PRA and broader nuclear safety communities regarding the maturity of Fire PRA, the degree of testing of methods and approaches, the degree of conservatism (compared to internal events PRA), and the consistency of results with operational experience. Unrealistic Fire PRA results could, of course, affect fire safety-related decisions and improperly skew comparisons of risk contributions from different hazards. In addition, concerns with realism can impact the perceived value of Fire PRA information by stakeholders. The issue of realism was discussed by presenting some preliminary comparisons of PRA quantitative and qualitative results with operational event data. (Work on the topic is still ongoing.)

The paper presented a number of summary observations, that maturity and realism are separate concepts, that Fire PRA is sufficiently mature to support major decisions, that the plant level statistical analysis performed to date does not support claims of conservatisms, that changes in relative core damage frequency (CDF) estimates indicate recent fire results could be conservative, that qualitative comparisons of actual Fire PRA scenarios look generally reasonable and that Fire PRA technology improvements are underway. It was also recognised that the needs of a risk-informed application can affect analysis realism.

In the discussion following the presentation, it was noted that the issue of realism is always present, and important to recognise when performing risk analysis – what is the pragmatic, application-specific balance between conservatism and realism in allocating analytical resources and how does this affect ultimate decision making.

The second paper was presented by Roman Grygoruk (AREVA NP, Germany) on the topic "*Fire Ignition Frequency Estimation: Component-based versus Room-based Approaches*". The two approaches were briefly described and a comparison was presented. The room-based approach is also known as the *Berry Method* (per the author of NUREG/CR-0654 [39], which describes the approach) and has been applied by a number of member countries since the end of the 1970s. The method uses information about factors of importance for ignition, spread, detection, extinguishing etc. to distribute a plant or building fire frequency over different locations in the plant. The absolute fire ignition frequency for the building is taken from plant specific operational experience or generic data sources. Plant walk downs are performed to support the collection of data on fire loads and ignition sources. The more recent component-based approach (from NUREG/CR-6850 in 2005 [5], [12], [13]) bases the assessment of fire frequencies for different ignition sources. The ignition sources considered include components as well as sources of hydrogen fires, transient fires and fires caused by welding and cutting. The conclusions from the paper are that both methods can be viewed as Top-Down approaches, the component-based approach is more flexible with regard to plant types since it starts from components, and that it does not require plant walk-downs in the way needed in the Berry Method and thus is more useful for new NPPs and in the design stage. It is further considered more precise with regard to ignition frequency and therefore allows for increased realism, The general conclusion of the paper is that component-based approach is preferred to the room-based approach as being more generally applicable, being more realistic, and also possible to use for new designs.[3]

---

3 It should be noted that the current "component based" approach of NUREG/CR-6850 allocates an assumed constant total fire frequency for a given component type (e.g., electrical cabinets) across all plant components of that type. Work is ongoing to develop the information (e.g., actual component counts) needed to develop a truly component-based approach.

One post-presentation observation was that to ensure the usability and reliability of the fire ignition frequency estimates, it is important to use consistent definitions for the events. Inclusion of subsequent, post-ignition events in the fire scenario, such as self-suppression, in the frequency data should be avoided as much as possible. (This viewpoint tends to favour the NUREG/CR-6850 approach. However, it should be recognised that as a matter of current practices, the room-based approach is still used.)

It should also be noted that even if the component-based approach does not necessarily require a plant walk-down, it is prudent to perform one. Even in the case of a design stage analysis, a mental walk-down of the design, which is also likely to be performed as part of a deterministic fire hazard analysis, can be performed.

The third paper was presented by Florian Berchtold (BAM, Germany) and the topic of it was "*Probabilistic set of filter criteria in the frame of Fire PSA*". Filter or screening criteria are used to successively focus the analysis on what is risk dominant. This is in line with the application of the graded approach required by WENRA and IAEA. The specific application of filter criteria as presented in the paper is to reduce the number of compartments needed to be analysed in more detail. The specific criterion used in Germany is based on fire load density (combustible energy content per floor area of the compartment). Compartments with a fire load density below 90 MJ/m² can be screened out. The paper states that the justification of the particular value of 90 MJ/m² is not well documented, nor does this criterion take into account varying compartment characteristics such as ventilation conditions, physical and chemical properties of the fire load as well as compartment sises. This was the background to develop a probabilistic set of filter criteria to address the limitations of the existing fire load criterion. The new method combines a deterministic model for the component failures with a four-dimensional sensitivity study on fictitious compartments enabling fast evaluation of room-specific conditions, and a probabilistic part that takes into account the model and parameter uncertainties in the comparison against limiting conditions. Cable temperature was used as a deterministic failure criterion but the method could be applied for other criteria as well. Using this approach, a compartment can be screened out in Fire PSA if the predicted failure probability is below a predefined accepted threshold value.

It is noted that the method is currently at a theoretical stage, and the paper also has some proposals for further improvements, e.g. more simulations, more data and additional criteria for other influences. Several other presentations also discussed this subject with regard to systems, structures and components (SSC) vulnerability in case of fire, and at what temperature or heat release rate (HRR) they are expected to fail.

It should also be noted that the presentation indicated a systematic, non-conservative bias (on the order of 20 %) in the predictions of FDS, which was used by the author to perform demonstration calculations. Such a bias would have led to non-conservative screening in a deterministic analysis. The assertion of temperature under-prediction was challenged by other meeting participants, who referenced past FDS verification and validation activities that showed a conservative bias on the order of 10 %. This discussion highlights the importance of: treating the uncertainty in model predictions (which could vary with the situation being modelled), the need for data to support assessments of this uncertainty, and the need for careful interpretation and use of such data.

3.3 Session 2 – Progress in Fire PRA methodology – Tools and data

Session 2 involved presentations covering topics ranging from automated tools for Fire PRA (including simulation models) to databases and data systems that could support Fire PRA.

The first presentation on a "*PSA Automatisation Tool*" by Albert Malkhasyan (TRACTEBEL, Belgium) highlighted the demanding complexity of modern Fire PRAs (as per the guidance of NUREG/CR-6850) and the associated requirement to handle very large amounts of data. These challenges make Fire PSA difficult to manage and time consuming, and motivated the development of an automated tool. The tool is linked to the PSA model used for internal events PSA and includes data processing from the input databases up to the final quantification of Fire PSA event trees. It includes simple fire models to perform scoping-level assessments of a fire's zone of influence, and also implements the human reliability analysis approach described in NUREG-1921 [18]. The tool is aimed at supporting quick turnaround, sensitivity and update analyses.

The second presentation titled "*Simulation Toolbox Development for Fire PRA*" was given by Simo Hostikka (Aalto University, Finland) is based on experimental and numerical simulation methods developed in the recent past for NPP Fire PRAs. The overall aim is to form a complete chain of simulation tools starting from fire development (in particular for liquid pool fires and cable fires, which represent the most frequent types of fire sources in such facilities), and ending with SSC and human response. The specific focus of the presentation was on developing new methods for the performance of fire barriers (including considerations of such things as availability, not just their thermophysical behaviour) for prevention of fire propagation from the fire compartment to adjacent ones, which represents the next step in the simulation chain. In this respect, a procedure for calculating the barrier failure probability by combining the stochastic fire simulations and the barrier response analyses was developed based on a new tool that crosslinks CFD fire simulation and finite element analyses of structures. Other features are still under development. These include the modelling of smoke induced failures of electronic equipment and of the stochastic human and organisational response to the fire. The latter will be implemented using an agent-based modelling approach instead of a more rigid flowchart (pre-defined network modeling) approach. Additional enhancements envisioned include a validated sub-grid scale model for cable objects and the associated heat transfer processes, and an extension of the structural analyses tool to facilitate modeling of fire barrier elements. Future work also includes the collection of user experiences in modeling fire effects in real NPP structures, and the development and incorporation of basic models for human decision making processes for simulating situations with limited instructions and procedures.

In the discussion following the presentation, the author noted the importance of experimental data to support the modeling effort (e.g. for the treatment of soot and its effect on electronic components), the large amount of resources required for such work (which inhibits solo work by a small organisation or country), and the value of work performed in the U.S. (particularly by the US National Institute of Standards – NIST), much of it sponsored by NRC. The author was also informed of other efforts to develop agent-based models for human behaviour, and the difficulties in validating such models.

The presentation "*Building up A Cable Management System and its Use for Fire PRA*" by Andreas Hempel (KABTec, Germany) provided information on the basic cable data needed for Fire PSA. Cables represent both a threat vector (being potential ignition and fuel sources – cables constitute the highest fire loads in NPPs) and a safety-related target. A cable management system provides a systematic means for documenting and accessing the characteristics of the different types and amounts of cables, their routing, and the electrical functionality and potential circuit failures modes in case of any initiating event, such as fire. A suitable and effective cable management system therefore reduces the work load for PSA analysts and assists the Fire PSA specialists in systematically considering cables, their failures characteristics and estimating the resulting damage frequencies under different boundary conditions. Once such as system has been installed in a NPP, it is a tool closing knowledge gaps with respect to cables and reducing uncertainties. The presenter noted that such a system identified a cable routing error at an existing plant, and that the implementation of such a system at an existing plant could involve around 2 people and 8 staff-years of effort.

The fourth paper "*Failure Mode and Effect Analysis of Cable Failures for Fire PSA with Quantification of Failure Mode Probabilities*" by Joachim Herb and Ewgenij Piljugib (GRS, Germany) presented a recently developed methodology for analysing fire induced cable failures and circuit faults within Fire PRA by means of a detailed and automated failure mode and effect analysis (FMEA). The approach is based on a database to systematically assess potential effects of cable failures caused by fire, while the probabilities of the different failure modes identified are quantified using the correlations given by NUREG/CR-6850 [5], [12], [13]. The cable failure mode probabilities (e.g. for hot short, fault to ground) are assigned to the corresponding probabilities of component failures (e.g. signal fails low or high), which are correlated by mapping to basic events in the Fire PSA model. The cable routing information (e.g., such as that provided by a cable management system described in the preceding paper) is stored in the cable database for all of the cables in the NPP. This enables the analyst to consider the effect of fire induced cable failures in the calculation of the core or fuel damage frequency by combining compartment specific fire frequencies with probabilities of failure modes and component effects respectively. The presentation noted the difficulty in defining "worst case" scenarios, and the importance of treating failures associated with control-related process feedback signals (which play a prominent role in highly automated plants). The presentation also provided a useful summary of estimated likelihoods of different fire induced failure modes. These do not include spurious actuation of such components as motor-operated valves, due to the infrequent use of multi-conductor control cables in German plants.

In the discussion following the presentation, it was noted that the results of the work might be useful for flooding and accident management applications. (The author noted that the former application would likely require an extension of scope to cover such things as junction boxes.)

Finally, the paper "*Extension of the German Database of Plant Specific Failure Rates for Fire Protection Systems and Components*" by Burkhard Forell (GRS, Germany) provided recent data on technical reliability of active fire protection features in various types of NPPs of different plant generations. These data are needed for event tree analyses, particularly for estimating branch point probabilities in the fault trees for Fire PSA. While the calculated failure rates can be used by the plants that provided the raw data (Fire PRAs for these plants will be carried out in the future as required by the German regulations [40]), the generic data are applicable, at least as a-priori information, in Fire PRA for any installation with active fire protection components of the same type or at least very similar characteristics.

Following the presentation, questions were raised concerning some of the results. In particular, one participant suggested that, based on his field experience, the probability of a fire door being open was on the order of 10 %, as opposed to the 0.1 % reported in the paper. The author's view was that even a value of 1 % was too high. This discussion indicates potential differences in member country practices and highlights the need to take such differences into account when using generic estimates.

3.4 Session 3 – Use and application of fire event databases

Session 3 focused on the various fire event databases that currently support national and international Fire PSA efforts. The presentations demonstrated useful applications and insights that can be supported by the creation of a robust database of fire events. As discussed in Session 1, it was observed that fire frequency estimation methods vary from a plant area/room-specific approach to a component-based approach. The general trend is for countries to adopt the component-based approach and this approach is supported by ongoing work being performed by both EPRI and the international community. The papers also show that the information collected by the database projects can support the estimation of other Fire PRA parameters. For example, they can provide empirical evidence to address a long-standing Fire PRA concern: the likelihood that a given fire presents a severe thermal challenge to potential targets [41].

The first presentation "*Fire Events Database – Insights and Opportunities from the Fire Event Database*" by Ashley Lindeman (EPRI, USA) focused on the updating of the EPRI Fire Events Database (FEDB) and potential applications that a more comprehensive database can support. The current database contains records for events occurring during the period 1968-2009. Reporting thresholds were briefly discussed as well as the definititions of the four main classifications of fire events contained within the new database structure, i.e. challenging (C), potentially challenging (PC), not challenging (NC) and unknown (U), whose distinctions are based on the induced fire environment (but not the plant effect). This topic led the participants to a brief discussion pertaining to the need for a consistent reporting threshold among all countries in order to create a database of fire events which can serve as a common souce for all users. The presentation also showed the ability of the database to support a more refined event binning structure and therefore improved analyses of potentially risk-significant scenarios (e.g., electrical cabinet fires, addressed as a broad group in Bin 15 of NUREG/CR-6850). The presentation noted that in the future, it is intended that fire event data will be collected by the Institute of Nuclear Power Operations (INPO), and that EPRI will update key parameter estimates derived from these data(e.g., fire frequencies, non-suppression probabilities) every 3-5 years.

In addition to the EPRI FEDB activity, a separate OECD fire events database project is currently underway. The main purpose of this database, as outlined in the presentation of the chair of this project, Marina Röwekamp (GRS, Germany) in her presentation "*Use and Applicability of the OECD FIRE Database Project*", is the collection of fire event data from 12 member countries. These data are used to estimate fire frequencies, analyse human performance, estimate fire supression response times, and investigate root and apparent cause information for use in PSA. This presentation focused on the ingishts that have been gained through member country co-operation as well as how the database is being used to help direct future research programmes, most notably the NEA High Energy Arcing Fault (HEAF) testing program [30]. Limitations of a broad data collection project were also discussed and country specific reporting thresholds again became a topic of discussion. The FIRE Database contains data from several member countries reporting all fire events as well as data from other member countries which only report the more challenging fire events. For example, although the U.S. has submitted the majority of event reports to the project, these reports are only for fire events which meet the reporting criteria of 10 CFR 50.72 (immediate notifications) and 10 CFR 50.73(Licensee Event Reports – LERs).

In the post-presentation discussion, workshop participants expressed interest in analyses showing if different rooms can (and should) be pooled, and in an exploration of potential organisational and procedural influences.

The focus of the presentation "*First Investigations Regarding Combinations of Fires and other Events in the OECD FIRE Database*" by Heinz-Peter Berg (BfS, Germany) was on one of the key items that was an area of broad interest during the meeting - combinations of events, including: a) fires and consequential events (e.g., secondary fires, explosions) induced by a fire, b) events (e.g., natural hazard events) and consequential fires induced by an event, and c) fires and nearly simultaneous but independent events. It was noted that in current Fire PSA the starting point of an analysis is a fire in one plant location. This investigation attempted to use the OECD FIRE Database to analyse fire events which behaved in a more complicated manner, potentially outside the current scope of Fire PSA guidance, as for example, seismically-induced fire scenarios or fire scenarios which caused unintended flooding situations through fire fighting activities. At the time of the analysis 45 out of a total of 415 fire events were observed to have some combination effects. Some of the events had minor impacts, others led to the loss of safety trains. This type of empirical analysis is valuable, as the impact of event combinations on plant response and overall plant risk are not well understood, and practical methods for assessing the risk significance of such combinations are still in the development stages.

One important finding of this session with regard to the use and application of fire event databases is that the approaches for estimating fire occurrence frequencies within Fire PRA strongly vary. Some analysts use generic frequency data while others mainly apply plant specific data. Two approaches are widely used, area or compartment based ones versus component based ones.

3.5 Session 4 – Fire safety analysis and research

The presentations of this session covered four important phases of a safety-significant fire scenario, namely ignition, fire development, fire detection and electrical circuit response. The work reported in the presentations is supportive of deterministic as well as probabilistic analyses.

In the first presentation of the session "U*.S. NRC Fire Safety Research Activities*" Nicholas Melly (NRC, USA) provided an overview of a number of ongoing research projects. The first experimental campaign that was presented aims at improving the realism of the electrical cabinet HRR distributions provided in NUREG/CR-6850 [5], [12], [13]. The fire loads within the cabinets have been defined according to the experiences gained during several site visits. It was noted that simple cabinet classifications (e.g., "protection system cabinets") may not be very good indicators of fuel loading. Both gas burner and liquid acetone fires are being used as ignition sources; the HRRs of these sources are smaller than those used in the NUREG/CR-4527 [42] tests cited in NUREG/CR-6850. Recognising that the distribution of experimentally measured HRRs is a direct reflection of the test matrix but does not necessarily reflect the relative frequency of relevant plant conditions, the experimental results will be used by an expert panel to develop HRR distributions useful for Fire PRA purposes.

The second experimental campaign, carried out as an international (OECD) co-operative activity, will focus on High Energy Arcing Fault (HEAF) events. The current NUREG/CR-6850 guidance for treating such events is based largely on a single, well-documented event (San Onofre, 2001). The intent of the programme is to quantify the phenomena (e.g. duration and energy content) and possible consequences associated with the event, its zone of influence and the subsequent fire. The experimental programme design is based on insights from investigations from an NEA WGIAGE task on High Energy Arcing Fault Events (HEAF) (cf. [29]) and results from the operating experience with HEAF fire events in member countries as documented in the OECD FIRE Database [36] and [52]. The experiments are ongoing; the corresponding report of the OECD experimental programme is expected in early 2016.

The third campaign deals with the effectiveness of incipient fire detection technology. The goal is to provide supplemental guidance for the treatment of very early warning fire detectors in Fire PRA using information collected from site visits and the results of several experiments (laboratory- and room-scale).

In his presentation, titled "*Fundamental Investigation on Successive Fire Due to the High Energy Arcing Faults Event for the High Voltage Switchgears*", Koji Shirai (CRIEPI, Japan) reported results from recent HEAF experiments. HEAF durations and arc energies were reported along with the possible fire ignition. The tests involved 6.9 kV and 8.0 kV switchgears; 480 V switchgear tests are planned. The arc energy threshold value of 25 MJ was found for the fire ignition. The presentation noted that a number of the tests did not result in fires, while others resulted in fires a few minutes after the HEAF event. A number of the tests also showed heavy mechanical damage to nearby cable trays and some tests generated missiles (in the form of broken bolts) that were found up to 15 m away. The cabinet pressure was investigated both experimentally and by numerical simulations using AUTODYN software.

The discussion highlighted the need for better understanding of the actual fire ignition process.

In the third presentation of the session, Nicholas Melly (NRC, USA) discussed the use of expert panels to interpret cable damage data and models and develop information useful for Fire PRA. The presentation title was "*Expert Judgement: An Application in Fire Induced Circuit Analysis*". The role of the experts in such a panel was to identify the failure modes leading to spurious operation and to assign conditional probabilities for them. Consensus results were obtained first by a Phenomena Identification and Ranking Table (PIRT) panel for the cable failure mechanisms. Next, following the guidance of NUREG/CR-6372 [43] regarding the use of expert judgement,[4] distributions were developed for the spurious operation conditional probabilities (assuming fire induced cable damage) and durations. These distributions were intended to represent the state of knowledge of the informed technical community (and not just the consensus of the panel members). Uncertainties were found for instance in cable ageing effects. Overall, using expert panels for creating a synthesis from complicated experimental observations, thereby transforming research results into Fire PRA application guidance, was found to be effective.

Next, Topi Sikanen (VTT, Finland) gave a presentation on "*Predicting the Heat Release Rate of Liquid Pool Fires Using CFD*". He discussed the development and validation of the liquid evaporation sub-model of the Fire Dynamics Simulator (FDS) software. The challenges of determining the mean absorption coefficients for the liquids were discussed. The first set of validation exercises showed that the accuracy of the predictions for the maximum burning rates was comparable to the empirical, experimentally-based correlations in the literature. Next, the influence of the absorption coefficient determination method and the internal liquid convection modeling on the pool burning dynamics was demonstrated by comparison against two sets of liquid evaporation rate data. The author indicated that although the work addressed open pool fires, the models also perform reasonably well for under-ventilated fires.

The final talk of the session was given by Anna Matala (VTT, Finland) titled "*Predicting Fire Spread of A Cable Tray Using Methods of Pyrolysis Modeling*". In her presentation, she introduced the steps of the pyrolysis modelling process: sample preparation, small and bench scale experiments carried out at VTT, model preparation and parameter estimation, and validation. The final set of validation cases consisted of the cone calorimeter and radiant panel experiments of a PVC cable within the U.S. NRC CHRISTIFIRE test series [23]. Simulations using FDS reproduced well the cone calorimeter tests at heat fluxes other than the one used for parameter estimation, but mixed results were obtained for the radiant panel tests.

In summary, both, experimental data and the predictive simulations for fire HRR – one of the most important yet uncertain fire characteristics – have been developed. Predictive simulations using CFD to develop HRRs from first principles are becoming more practical. More work is needed to extend the predictive capabilities of such models on a wider scale of fuels and to validate them in ventilation controlled conditions. The importance of the multi-scale data on material behaviour was clearly highlighted. It was proposed that experimental capabilities to characterise the NPP fuels in terms of advanced model input parameters should be developed. In the discussion following the presentations, it was suggested that such activities should be included in all experimental campaigns dealing with fire ignition and spreading. Opinions were also presented that the risk significance of international cable qualification concepts could be critically examined.

The main characteristics of HEAF events are being investigated by experiments. However, the process of fire ignition after the HEAF – see [29] and [30] – is not yet well understood at the level of physical and chemical processes. More research on those effects would likely be informative.

From the viewpoint of the research result dissemination, the U.S. NRC experiences on the use of expert panels in the analysis of experimental data towards practical guidance were encouraging. Such a

[4] The NUREG/CR-6372 approach to the use of expert judgment is commonly referred to as the "SSHAC approach," as the report was authored by the NRC's Senior Seismic Hazard Analysis Committee. Although the report was developed to support seismic PRA applications, its principles and use are much broader.

systematic approach could also be applied in the other topics of fire safety assessment, such as the integration of deterministic simulations into the probabilistic framework.

3.6 Session 5 – Requirements and guidance for Fire PSA

This session focused on the work of the regulators in the area of Fire PSA as well as on international activities in the area of PSA guidance. Four papers were presented in Session 5, three of them discussing national enhancements in Fire PSA guidance and another one presenting the IAEA position in how to expand Fire PSA scope and ensure sufficient quality of such assessments.

The first presentation on "*Insights on Recent Regulatory Guideline of Fire Protection in Finland*" by Matti Lehto (STUK, Finland) provided insights on the most recent regulatory experience from the application of national fire safety guidelines. Finland has a very detailed set of guidance documents and upgraded its safety regulations in December 2013. The upgrade included issuing guidance for PSA level 1 and level 2 including, in those, the need to address fire protection. The guidance includes many features derived post-Fukushima such as defence in depth and combinations of events.

In relation to fire, the new requirements for defence in depth include addressing both structural and active fire protection. For combinations of events, the effects of fire protection activities on nuclear safety are also expected to be addressed (e.g., flooding as a result of fire extinguishing activities, recognising that post-Fukushima increases in the fire water system inventory may increase the flooding risk).

An interesting aspect presented was the investigation of the reliability of the water system not only with respect to fire protection but also for accident management needs.

Usha Menon (CNSC, Canada) gave a presentation on the topic "*Regulatory Framework and Insights from Fire PSA of Canadian Nuclear Power Plants*". As with Finland, a review of regulatory guidance and requirements has been undertaken. There are significant new revisions that will be issued soon. Canada requires Level 1 and Level 2 PSAs to include internal and external hazards for full power as well as for low power and shutdown plant operational states. Fire PSA (performed by the licensees using the methods described in NUREG/CR-6850 [**11**], [**12**] and [13]) is used to identify weaknesses in the defence in depth concept and to evaluate benefits of plant modifications. An example presented was an improvement in safety that was achieved following the issuance of a revised Fire PSA for the Point Lepreau reactor. (The upgrade involved a design change to reduce the spread of oil fires in the reactor building.) It was further explained that for Canada the results of recently performed Fire PSAs indicate that fire events contribute approximately 30 % to the overall CDF for internal and external events.

In the "*Guidance for Performing Fire PRA in Germany*" by Heinz-Peter Berg (BfS, Germany), the most recent developments regarding the use of and requirements on probabilistic fire assessment within the regulatory process were presented. Fire PRA is required in Germany as part of the PSA within Periodic Safety Reviews (PSR) and is a supplement to deterministic safety demonstrations. As a result, Fire PSAs have been conducted for all operating NPPs in Germany.

However, in the past, Fire PSA has been focused on full power plant operational states. For those plants that underwent a second PSR the scope was extended to low power and shutdown states for internal events but not fully to internal hazards.

New requirements have been recently issued and a Fire PSA for low power and shutdown states has to be performed for those plants for which a third PSR is required. This means that two plants in 2016 and 2017 will need to have a completed Fire PSA. Guidance on carrying out such analyses will be part of a document which is expected to be issued in 2015 [21]. This guidance document is hoped to be issued alongside updated nuclear safety standards on fire protection also requiring the analysis of combinations of fires with other events.

It is also important to note that within the *"Safety Requirements for Nuclear Power Plants"* issued in January 2013 [40] there is a statement that PSA shall be used in the future to assess the safety significance of modifications of measures, equipment or the operating mode of the plant for which a significant influence on the results of the PSA can be expected.

Thus, in addition to the requirements to perform a Fire PSA within PSR, this procedure also has to be applied for modifications with respect to fire protection in the intervening period between PSRs.

Finally, a presentation titled "*Fire PSA Attributes in the Integrated IAEA Guidelines for PSA Quality*" was given by Artur Lyubarskiy (IAEA, Austria). The IAEA TECDOC on PSA quality issued in 2006 (TECDOC-1511 [**44**]) has been successfully used by several IAEA Member States. However, this document only covers internal events at full power. Therefore, an expansion of this TECDOC is under preparation which will include internal and external hazards and low power/shutdown plant operational states. The additional topics will be integrated in the existing structure of the TECDOC providing general attributes and special attributes. These attributes can be used to check if the PSA under investigation fulfils the attributes and is in line with the required quality of a PSA. As compared with the ASME/ANS PRA Standard, the revised TECDOC is expected to provide: enhanced requirements (often Capability Category III) for specific applications (this is also a feature of the original TECDOC); a PRA-element focused structure (as opposed to a structure that separates requirements for fire, seismic, etc.); and a requirement to characterise parameter estimates with full distributions (rather than only mean values). This revised TECDOC may support the development of more comparable Fire PSAs in the future. A consultancy meeting is planned in May 2014 which should lead to a completion of the document by the end of 2014.

It is important to note that the document on fire safety in the operation of NPPs is included in the post-Fukushima list of updates identified by IAEA in its action plan.

The following observations can be made from the papers presented and the subsequent workshop discussion:

- The importance of Fire PSA for low power/shutdown plant operational states is growing. (Considering the likelihood of more and more reactors being permanently shut down, the assessment of the risk associated with fire induced loss of cooling of spent fuel in storage could be a special topic of interest.) Therefore, it would be worthwhile to exchange practices currently applied and identify good practice guidance for these plant operational states for Level 1 and Level 2 Fire PSA.

- Post-Fukushima investigations have identified the need to perform PSA of combinations of fires and other internal and external hazards (seismic, flooding, HEAF, explosions, etc.) in a systematic manner. It would be very helpful to share methods and good practices and experiences on these activities among member states.

The questions arising are:

- Does WGRISK need a task to seek good practice guidance on these?

- Is there a need to modify the 1999 SOAR to incorporate these topics?

- Would such efforts duplicate work that is being carried out by the IAEA or others?

3.7 Session 6 – Insights from probabilistic Fire analyses (PRA)

The aim of Session 6 was to provide insights from recent Fire PRAs. The five papers presented in this session focused on electrical cabinet fire experiments and fire simulation, human reliability analysis (HRA), fire induced spurious operations analysis, and sensitivity analysis.

The first paper "*Focus on the Studies in Support of Fire PSA of the French 1300 MW_e Nuclear Power Plants*" by Julien Espargilliere (IRSN, France) provided experimental and simulation results used to support a French Fire PSA. The experiments were performed to assess key parameters of fire simulations that employ IRSN's SYLVIA code (a two-zone model based on a ventilation network). As discussed in a later presentation by Fabienne Nicoleau (summarised below), SYLVIA is used to support the quantification of fire induced CDF.

The paper also discussed the results of experiments used to quantify key parameters of the simulations. Fuel source fire properties were defined by means of electrical cabinet fire tests in an open atmosphere under a large scale calorimeter to characterise heat release rate, radiant heat flux and combustion products. The cabinet fire tests were performed either with both open and closed cabinet doors.

Electrical components were exposed to increased temperatures to assess their thermal damage criteria. Additionally, damage criteria associated with the combined effects of soot concentrations and elevated temperatures were obtained. Despite the high amount of PVC involved in the cabinet tests, no damage due to corrosion was found. Three damage criteria of electrical equipment were considered in the simulations: gas temperature around the equipment > 65 ^{o}C; gas temperature > 95 ^{o}C; and gas temperature > 65 ^{o}C and soot concentration > 1.5 g/m^3. These damage criteria were met at 10, 17, and 14 minutes, respectively.

Fire spread to neighbouring cabinets was assumed to take place according to the approach given in NUREG/CR-6850 [5], [12], [13]. This resulted in estimated spread from the originating cabinet 15 minutes after ignition, spread to three cabinets are in 15 to 30 minutes, spread to five cabinets in 30 to 45 minutes, and spread to seven cabinets in 45 to 60 minutes. The maximum heat release rate in the experimental study was about 1.6 MW; this was reached at 12 minutes. The maximum heat release rate (HRR) was assumed to be the experimentally observed single-cabinet peak HRR of 1.6 MW.

The second paper "*Insights from HRA and MSOs Analyses in Spanish Fire PRA*" by Pedro Fernández Ramos (Empresarios Agrupados, Spain) provided experiences from the updating of HRA analyses and multiple spurious operations (MSO) analyses for two plants. The HRA update was based on NUREG-1921 [18]; the MSO analysis update was based on NEI-00-01 [**45**].

For the HRA, the performance shaping factors used consider the following issues in case of fires: Are the required indications available for the operators; are they accurate; are alternative information sources provided in the procedures; is training given for the operators for specific scenarios?

It was concluded that fire events affect instrumentation in many ways and the indications may not be accurate. The induced damage was specific to each fire area, but sheets showing instrumentation and equipment in each area were seen as adequate for supporting operator decisions in the case of conflicts in indications. Spurious alarms requiring direct operator response were considered separately, but no additional malfunctions were identified as a result of spurious alarms.

The presentation indicated that NUREG-1921 [18] provided a substantial improvement over previous guidance, but also that a significant amount of effort was needed to perform the analysis. The presentation also noted potential inconsistency with the methods used to perform HRA for internal events analysis. Overall, the analysis resulted in higher HEPs, higher assessed levels of dependency, and CDF increases.

NEI-00-01 [**45**] provides a list of MSO scenarios. The presentation indicated that an expert panel was used to review this list and map MSOs according to whether they affected either one train or different trains, affected alternative systems performing the same function, or induced scenarios not identified before. Regarding the last point, it is notable that the analysis did identify some new scenarios requiring treatment.

The third paper "*Uses of Fire PSA and Sensitivity Studies*" by Fabienne Nicoleau (IRSN, France) highlighted the Fire PSA studies of 1300 MWe reactors in France based on NUREG/CR-6850 [5], [12], [13]. Critical compartments were identified by means of component importance measures included in the internal events Level 1 PSA. The study also addressed neighbouring compartments. National operating experience covering more than 830 fire events during about 1400 reactor years was utilised to estimate fire frequencies and to define fire scenarios. The SYLVIA code was used to simulate fire scenarios and to assess the potential for component failures in the initial fire compartment and neighbouring compartments. The analysis developed an event tree for the fire scenario, a bridge tree for fire procedures, and a modified Level 1 event tree to address plant response. The fire risk was quantified using RiskSpectrum software.

Sensitivity studies were performed to address uncertainties in the damage criteria for electrical cabinets and the propagation of electrical faults affecting the power supply system. For the first case, the cabinet damage criterion was increased from 65 ^{o}C to 95 ^{o}C (for electronics and electrical equipment). This reduced the fire induced CDF by 92 %, due to reductions in the likelihood of fire induced initiating events and mitigating equipment failures. In a second case, the experiment-derived damage criteria presented previously by Julien Espargilliere (65 ^{o}C and 1.5 g/m^3 soot) were used. This led to a decrease in fire induced CDF of 48 %. In a third sensitivity study, assuming the loss of additional train(s) in case of a fire in a non-classified switchboard increased the fire induced CDF by a factor about 4 to 10 (as compared with the reference study).

These studies demonstrated the importance of key assumptions and the value of experiments in providing a basis for these assumptions.

The paper "*Fire PRA and Multiple Spurious Operations Study in Korea UCN 3 Unit*" by Kilyooo Kim (KAERI, Korea) presented a MSO study based on guidance given in NEI-00-01 [**45**]. Combinations of equipment failures leading to spurious operations were handled by fault trees. Possible spurious operations due to fires affecting associated cable circuits within individual fire compartments were then identified. In total, 193 components and 1907 cables related to these components were included in the analysis.

The analysis identified 17 scenarios and 40 possible compartments where fire could cause MSOs (. The typical effects of the scenarios induced by spurious operations included effects on seal cooling of the reactor coolant pumps, reactor vessel head vent valves, and atmospheric dump valves.

A mapping table was used to identify the affected components and their failure modes in case of fire in the compartment. IPRO-ZONE software was used to generate automatically the Fire PRA model by means of the mapping table and the internal PRA model. The treatment of an example scenario is included in the paper.

The presentation noted the difficulties in performing circuit analysis and the need for improved guidance.

The last paper "*Spurious Operation Scenarios - Knowledge Sharing from the MSO Mapping of Ringhals PWRs*" by Anna Bjereld (Ringhals AB, Vattenfall, Sweden) summarised an analysis of spurious operations to identify scenarios to be included in the fault trees of a Fire PRA.

Previously, the Ringhals Fire PRA assumed that fire induced spurious operations were unlikely (it assumed that fire damage would cause a loss of power or signal) and fires were not expected to result in a loss of coolant accident. However, cable fire tests performed by EPRI and NRC showed that spurious operations induced by fires are not unlikely. The presentation noted that these test results are still counter-intuitive to many people, and the misperception of low likelihood remains a challenge to be overcome.

The applied method was based on NEI-00-01 [**45**]. The mapping mainly covered the identification phase, screening phase, and verification phase. The identification phase resulted in a comprehensive, preliminary list of MSOs. The screening phase resulted in a plant specific draft list of MSOs; scenarios considered not possible in Ringhals PWRs were withdrawn. Deterministic screening criteria were applied in this phase. Probabilistic criteria were considered, but not applied although a risk-informed method is provided in NEI-00-01 [**45**]. (For the case of Ringhals, the deterministic method was seen as more important and applied to the work.) The verification phase, which was performed by an expert panel, resulted in a final plant specific list of MSOs. The outcome of mapping is the input needed for further fault tree analysis.

Based on the mapping of spurious operations, 25 relevant scenarios were identified. These which include spurious valve operations leading to loss of coolant. Most of the identified scenarios arise due to control circuit failures in relay rooms, cable rooms and switchgear rooms. Additionally, transmitter cable damages were identied as possible initiators for spurious scenarios. The fire induced spurious operation scenarios were included in the Ringhals PRA model.

The following main conclusions can be drawn from the papers:

- Electrical cabinet rooms are one of the most important target areas in Fire PRA. In case of fire, electrical cabinets are vulnerable to the combined effects of increased temperature and soot. Electrical cabinet failures can result in spurious operations, faulty information or loss of information for the operators, and possible loss of control of many safety systems. Due to the uncertainties in key parameters and in the viability of different modeling assumptions regarding electrical cabinet room fires, there are large uncertainties in the associated CDF estimates.

- Studies of MSOs are important as they can identify many significant scenarios. However, the mapping of fire induced spurious operations is a large and complicated task to perform. Good example cases are available, however there may be a need for improved guidance.

3.8 Session 7 – Fire PSA applications

Session 7 addressed a number of different Fire PSA analyses. The three papers presented in this session discussed:

- the risk associated with hydrogen fires in a turbine hall,

- the results of a Fire PRA update for Olkiluoto Units 1 and 2,

- the development of a cold shutdown Fire PSA at Forsmark NPP.

The first paper "*The Fire on Turbine Generator as an Important Fire Risk Contributor at NPP Dukovany (VVER-440)*" by Ladislav Kolar (UJV Rez, Czech Republic) presented the study of fires involving the oil system and the hydrogen-cooled generators of the turbine. For the hydrogen fire two types of scenarios were considered: scenarios involving only hydrogen combustion and scenarios involving a hydrogen explosion prior to the fire. The EPRI FIVE model was used to estimate fire effects, and a TNT-equivalent model was used to estimate explosion effects. The Fire PSA was improved by using the cable routing database in order to better take into account the location of power and control cables for safety-related components. This study showed that fires in the turbine hall of Dukovany are the dominant contributors to the overall fire risk and significant contributors to the total plant CDF (all initiators), and that good ventilation is the best provision for explosion prevention and consequent risk reduction.

The second paper "*Updating of the Fire PRA of the Olkiluoto NPP Units 1 and 2*" by Lasse Tunturivuori (TVO, Finland) presented the results of the updated Fire PSA conducted for the Olkiluoto 1 and 2 NPP units. The updating of this study provided a comparison of the fire frequency results estimated using a component-based methodology and a room-based methodology. (These approaches were also discussed by Grygoruk in Session 2.) For some rooms, housing components with high fire frequency, the component-based fire frequency methodology led to higher results. This was notably the case for rooms containing the main feedwater pumps, a large amount of electrical equipment (e.g., the relay rooms), and rooms with hydrogen systems. The Fire PSA showed that a cable tunnel fire affecting two out of four redundant trains (each of 50 % capacity) increased the conditional core damage probability by a factor of ten, as compared with a scenario in which only one redundant train is affected by the fire. The author noted that this design weakness finding, which underlines the significance of physical separation between safety-significant equipment and active fire suppression systems in rooms housing redundant trains, would not have been obtained without a Fire PRA. The author also noted the use of relatively simple models (e.g., fire spread probabilities based on engineering judgement, assumed loss of all components within a room should a fire spread). Overall, the Fire PRA update led to increases in CDF for many rooms, and increased the total CDF by about 18 %. The author indicated that the CDF estimates may be somewhat conservative, but also indicated the modeling conservatisms were understood, and did not feel they affected the usefulness of the PRA.

The third paper "*Experience from Developing and **I**mplementing Shutdown Fire PRA at Forsmark NPP*" by Erik Cederhorn (Riskpilot AB, Sweden) presented the methodology applied to the Fire PSA at Forsmark NPP for cold shutdown states as well as the corresponding results. This study showed that, due to the ongoing maintenance, the CDF induced by fire during cold shutdown periods is one magnitude higher than during power operation, and about 5 % of the total CDF. The analysis focused on fully developed fire scenarios involving the fire induced loss of residual heat removal. The analysis did not address fire induced spurious operations, but assumed a fire would lead to the loss of all electronics in a room. The analysis employed judgement-based increases in human error probabilities to account for direct fire effects as well as possible route interdiction.

The applications of Fire PSA presented in this session illustrated practical analyses for such technically challenging problems as the treatment of combined hazards (e.g., fires and explosions), cable tunnel fires, and shutdown conditions. These analyses demonstrated that useful results can be obtained despite the modeling simplifications needed to generate these results.

3.9 Session 8 – Extension of Fire PSA scope and applications

The last session focused on Fire PSA applications extending beyond typical practice. Two papers were presented, the first one concerning Level 2 Fire PSA, the second Fire PSA for the post-commercial safe shutdown phase.

The first paper "*Latest Extensions of the Loviisa Fire PRA*" presented by Antti Paajanen (FORTUM, Finland) discussed the modeling and software challenges in extending the plant's Level 1 Fire PSA to analyse the potential impact of fires on severe accident sequences. The focus of the study was on system impacts; potential phenomenological interactions (e.g., the effects on a release) and possible effects on operator actions are being thought about but have not yet been analysed. The results of the study shows that only a minor portion of the fire induced initiating events results in a failure of the containment and a large radioactive release. This is mainly due to the fact that the Severe Accident Management (SAM) systems are mainly located in the reactor building, where the fire frequencies are relatively low, and are powered by a separate SAM diesel system isolated from other diesel generators.

The second paper "*Conducting Fire PSA for the Post-commercial Shutdown Phase*" by Michael Türschmann (GRS, Germany) gave first insights on the extension of the Fire PSA approach from power operation to low power and shutdown states including the post-commercial safe-shutdown phase. In particular, during that phase the reactor is de-fuelled and most major systems are out of service, empty, depressurised, and cold; fuel damage frequencies of interest involve fuel elements that are stored in the spent fuel pool. The plant modifications being considered in the plant specific application include a reduction in the number of emergency diesel generators, the movement of an independent emergency cooling water system, and the modification of the ultimate heat sink for residual heat removal. In addition, the German approach tries to consider as far as possible not only fires as singular events but also event combinations of fires and other plant or site specifically anticipated events.

The presentations in this session demonstrated that Fire PSA methods and tools can be used in practical analyses for extended (beyond Level 1, at-power) applications.

3.10 Session summaries and facilitated discussions

The final session of the workshop involved presentations by the chairs of the technical sessions summarising key points from their sessions followed by facilitated technical discussion.

The Session 1 ("Progress in Fire PRA Methodology - Relevant Issues and Methods") summary was presented by Per Hellström (SSM, Sweden). Key viewpoints and questions raised during the summary and discussion include:

- There are differences between deterministically-oriented and Fire PRA practitioners. This can lead to different visions of the purpose and practice of Fire PRA for different technical groups and cultures. However, it should be recognised that a risk-informed process includes deterministic as well as PRA analysis.

- There are questions as to whether fire-specific acceptance criteria are needed. Problems can arise when using PRA results to show that plant operation is safe. PRA is best used to improve plants.

- When estimating fire frequencies, should the analysis distinguish between energised and non-energised components?

- Some parts of current Fire PRAs are very refined (e.g., the treatment of spurious operations), whereas other parts can use very broad assumptions (e.g., a severe fire leads to complete burnout of a compartment). Should there be an aim for a comparable level of analysis?

- Although a number of presentations identified simplifying assumptions in performing Fire PRA, none of the workshop participants indicated that there were any problems with analysis realism sufficiently severe to prohibit the use of Fire PRA in practical decision support.

The Session 2 ("Progress in Fire PRA Methodology - Tools and Data") summary was presented by Marina Röwekamp (GRS, Germany). Key viewpoints and questions raised during the summary and discussion include:

- There appears to be a trend in increasing the use of automatic PRA model construction methods and tools. (Examples were illustrated during the workshop but the trend is broader than Fire PRA.) It may be interesting to have a WGRISK discussion on this topic.

- It would be useful to have automatic tools to support circuit analysis, possibly with direct links to fire models.

The Session 3 (“Use and Application of Fire Event Databases”) summary was presented by Nicholas Melly (NRC, USA). Key viewpoints and questions raised during the summary and discussion include:

- Updated fire event databases can support improved modelling and realism (e.g., through the parsing of the single NUREG/CR-6850 bin for electrical cabinet fires - Bin 15, data to support an increased focus on HEAF events).

- It should be recognised that different countries employ different reporting criteria for fire events, and this leads to heterogeneity in international database activities such as OECD FIRE. Is there a desire to develop and document best practices on reporting levels?

- It would be useful to use the databases to explore combinations of events (e.g., fire plus consequential events, initial events plus fire).

- More work can be done to refine the fire event data to provide additional information (e.g., detection and suppression times) and also refine uses (e.g., moves to a component-based ignition frequency model).

The Session 4 (“Fire Safety Analysis and Research”) summary was presented by Simo Hostikka (Aalto University, Finland). Key viewpoints and questions raised during the summary and discussion include:

- HRR is the most important parameter; experiments and numerical modelling studies are ongoing for the HRR determination and prediction. Examples in the workshop included liquid pool fires and electrical cabinets and cables.

- Multi-scale experiments are important in the characterisation of combustible materials for the determination of fire model inputs and model validation.

- The risk significance of international cable qualification concepts should be examined.

- One important mechanism (HEAF) is just starting to be explored. The current results can already be used to estimate the zone of influence and the means of protection. The actual mechanism of HEAF-induced fire ignition remains unknown and requires basic research.

- Expert panels have proven useful for transforming experimental data results into practical Fire PRA inputs. Systematic uses of expert judgement in other portions of Fire PRA (e.g., in integrating probabilistic simulations into the PRA framework) might also be useful.

The Session 5 (“Requirements and Guidance for Fire PSA”) summary was presented by Neil Blundell (NEA). Key viewpoints and questions raised during the summary and discussion include:

- As part of a general PSA trend, there is wide use of Fire PSA, including extensions to low power and shutdown modes of operation. Even countries with deterministic regulatory approaches (e.g., Germany) are making increasing use of PSA (e.g., to evaluate plant modifications as part of PSRs).

- Post-Fukushima requirements include the consideration of defence in depth and combinations of events.

- Should WGRISK pursue the development of good practice guidance?

  - It should be recognised that the European Union's ASAMPSA_E Project is underway; should WGRISK wait to see the outcome?

  - Other countries (e.g., Sweden) are updating their regulatory requirements (e.g., to identify combinations of events that need to be addressed in safety cases).

  - Caution is needed when developing best practice guidance for a field in the development stage.

  - When developing guidance, it should be recognised that more and more plants are shutting down.

The Session 6 ("Insights from Probabilistic Fire Analyses") summary was presented by Matti Lehto (STUK, Finland). Key viewpoints and questions raised during the summary and discussion include:

- Fire-induced failure criteria are multi-dimensional; synergistic effects between phenomena (e.g., heat and smoke) should be considered.

- Better analysis methods and tools are needed for HRA and MSO analysis.

The summaries of Sessions 7 ("Fire PSA Applications") and 8 ("Extension of Fire PSA Scope and Applications") was presented by Remy Bertrand (IRSN, France). It was noted that the workshop has involved considerable discussion on Fire PSA methodology and results. Nevertheless, needs were expressed for

- studies of fire induced spurious operation,

- extending the analysis scope to address low power and shutdown modes (with particular attention to decommissioning plants),

- using the results of Fire PSA in risk-informed decision making, particularly when the uncertainties are important, and

- addressing the effects of smoke and heat on digital instrumentation and control.

Finally, in her closing remarks, the workshop chair (M. Röwekamp) indicated that the workshop:

a) showed the value to Fire PRA of insights from fire-related operating experience, and

b) provided valuable material on performing Fire PRA not only for operating nuclear power plants, but also for new reactors at the design stage and for plants under decommissioning.

3.11 Post-Workshop discussion and writing session

Following WGRISK best practices in workshop conduct (cf. [46]), a post-workshop session was held immediately following the workshop to discuss the workshop outcome and develop an initial draft of the workshop proceedings, including workshop conclusions and potential WGRISK recommendations. As indicated in Section 1.3, this session involved the session chairs and members of the WGRISK task core group.

Key points raised during this session are as follows.

- There was considerable discussion during the workshop on Fire PRA for shutdown modes of operation. The fire risk during shutdown may be significant, and the need for analysis may be increasing with the increased number of plants entering decommissioning. It was recognised that there may be differences between analyses for operating and decommissioning plants. There may be a need to identify best practices in current analyses of shutdown conditions and to explore changes in current methods (which were developed to address at-power conditions).

- The treatment of multiple events and multiple hazards (e.g., seismically-induced fires, turbine building fires and explosions) remains a technical challenge. It would be useful to identify countries that are making significant efforts in this area.

- Level 2 PSA challenges include the treatment of fire effects on containment isolation (including consideration of MSOs) and on post-core damage accident development and release, and the treatment of associated operator actions in the HRA.

- Recognising that budget constraints can limit the performance of Fire PRA, it is important to communicate plant improvements based on PRA results. (Even intermediate PRA results routinely lead to plant changes that improve plant safety.) Furthermore, as with PRA in general, the process of performing a Fire PRA can lead to improved understanding of the plant, which is a benefit in itself. It could be useful for WGRISK to take a role in such communication, sharing lessons learned regarding PRA-based improvements possibly via a database of improvements developed from member country inputs.

- Options for the principal WGRISK post-workshop follow-up activities include the development of: a SOAR to update NEA/CSNI/R(99)27 (which was published in 2000, based on a member survey and a 1999 WGRISK workshop in Helsinki), a condensed SOAR, a Technical Opinion Paper (TOP) to update the WGRISK TOP on Fire PRA (which was published in 2002 [48]), and the development of a targeted Fire PRA guidance document or database. (Of course, combinations of these options could be considered.)

  – A SOAR provides a detailed, referenceable document that can answer country-specific questions (e.g., how do the country's methods, models, tools, data, results, and applications compare with those of other countries), and identify areas of agreement or disagreement across countries. However, the development of a SOAR (which often involves the development and distribution of a survey questionnaire and an analysis of the questionnaire answers) takes considerable time and resources and member countries might want to spend these on developing improved guidance.

  – A condensed SOAR, which might be drafted by a group of authors, could provide a summary of current Fire PRA approaches and an annotated bibliography for further reading. Given the broad international use of NUREG/CR-6850 and related guidance documents as illustrated in the workshop, a survey would not necessarily be required. A recent paper developed for publication in an upcoming revision of the Society for Fire Protection Engineers Handbook could be a useful model [47].

- A TOP, typically drafted by a lead author or set of authors and then commented on by the broader group, generally includes a high-level summary of the state of the art as well as the group's opinion on key points (e.g., the maturity, realism, and practical uses of analyses).

- A Fire PRA guidance document/database could address specific areas based on member country experiences. The areas could be identified through a targeted survey questionnaire that identifies where improved guidance would be beneficial. It could document member country lessons regarding successes and challenges in the application of current guidance documents (e.g., NUREG/CR-6850, NUREG-1921), recognising that, as discussed in the workshop, some member countries are selectively using portions of these documents.

- A potential long-term activity could involve the development of a Wiki page on Fire PRA. Such a page, which could be based on the condensed SOAR (with its annotated bibliography), could support the WGRISK vision of providing timely information (in this case, information not subject to the delays associated with a full report publication cycle) and serving as an internationally recognised, authoritative source. The viability of this activity would depend on the ability of the NEA's information technology resources to host such a page, as well as the resources of WGRISK members in maintaining the page.

- Some of the above options represent new types of activities for WGRISK. However, they have the potential for developing useful resources for practical applications by member countries in a more timely manner than conventional options.

4. CONCLUSIONS AND RECOMMENDATIONS

4.1 Conclusions

Based on the workshop presentations, discussions during the sessions (including the final and opening sessions), and the post-workshop discussion and writing session, it appears that Fire PRA has achieved a reasonable level of maturity. According to the practices in many countries, Fire PRAs are performed within a consistent framework and many employ common methods, tools, and guidance. There appears to be a common understanding of weaknesses and work is ongoing to bolster key areas. Most importantly, despite the recognised weaknesses, the results of Fire PRA are being used to support risk-informed decision making by plant owners and operators and by regulators.

The specific conclusions below are organised according to the workshop's objectives, which are stated in Section 1.2 of this report.

4.1.1 Status of Fire PRA Including Recent Developments

The workshop covered aspects of all elements of Fire PRA (ignition, growth and suppression, and plant response) as well as topics beyond the scope of many applications (Level 2, low power and shutdown operations including decommissioning). The discussion identified many commonalities as well as challenges.

Models, methods, tools and data

- There appears to be a general movement towards the use of a component-based (as opposed to a room based) approach for estimating fire frequencies.

- There are strong commonalities in international Fire PRA and related activities due to: the widespread use of common methods (notably those provided in NUREG/CR-6850 and associated documents), tools (e.g., the FDS code for detailed fire modeling), and fire experiment data (e.g., as developed by the CHRISTIFIRE programme); and international participation in the OECD FIRE Database Project.

- Some organisations are developing automated tools to support the practical implementation of current Fire PRA guidance, as the effort required for such implementation can be very high.

- A number of member countries are performing experiments to address gaps in empirical knowledge (e.g., regarding HRRs for various fires, HEAF effects, synergisms between smoke and heat as damage mechanisms).

- As the model and experimental uncertainties are being estimated, the basis and methods for treating these uncertainties within Fire PRA should be developed.

- The volume and quality of fire event data from operational experience is increasing. These data are now being used to address factors previously only considered through modeling and expert judgement (e.g., the likelihood of challenging fires) and to explore new topics (e.g., the frequency and characteristics of events involving multiple hazards, including fire).

- It was recognised that consistent reporting criteria among member countries would increase the usefulness of international fire event data collection activities.

Regulatory framework

- The use of Fire PRA (as well as PRA as a whole) is increasing, even for countries that employ a deterministic regulatory approach. In Germany, for example, Fire PRA is an important part of the PSR used to make a safety case for a plant.

- Following their consideration of lessons learned from the Fukushima Dai-ichi reactor accident, a number of countries have established new regulations to address combinations of hazards and events (e.g., the effects of fire extinguishment activities on SSCs).

- A number of countries also have requirements for Fire PRA addressing low power and shutdown conditions as well as full-power operations. Some countries require a Level 2 Fire PRA as well as a Level 1 analysis for such conditions.

Standards and guidance

- As indicated above, there is widespread international use of NUREG/CR-6850 and associated guidance. In a number of countries, these documents are supplanting guidance provided in NUREG/CR-0654.

- A number of participants expressed a need for additional or improved guidance to support key elements of Fire PRA, including circuit analysis and HRA.

- A number of international organisations and projects (notably, IAEA and ASAMPSA_E) are developing Fire PRA guidance that should be considered in any future guidance development activities.

Good practices

- Empirical data (e.g., from operational experience) can be used to support: a) the qualitative identification of potentially important fire scenario characteristics (e.g., as exhibited in major fire events), and b) the quantitative estimation of various Fire PRA model parameters (e.g., the likelihood of challenging fires, the reliability of fire protection SSCs).

- Expert panels can be useful to transform experimental results into practical inputs for Fire PRAs. These panels can address differences between the experimental and field conditions, can factor in other sources of relevant information, and can deal with the fact that a set of results generated to address an experimental test matrix does not, in general, constitute a random sample of direct use in a PRA. Practical guidance for using expert judgement in a structured fashion is available.

- Even when fully vetted Fire PRA tools are not available to address all of the technical challenges arising in new applications (e.g., low power and shutdown conditions, Level 2 analyses), useful insights (e.g., potentially effective risk management options) can often be gained using simple tools and simplifying assumptions.

Results and applications

- Confirming the results of previous studies, the workshop showed that fire remains, broadly speaking, a significant risk contributor for at-power operations, and appears to be significant even for shutdown states. Not surprisingly, important contributions come from electrical cabinet rooms and cable rooms. For some plants, turbine building explosions and fires can be dominant contributors to fire risk and significant contributors to overall risk.

- Fire PRA is being used in several countries to support plant improvements. It has been used to identify areas of weakness that would not be discovered (or at least would be difficult to discover) using deterministic methods.

- Fire PRA might also be useful in more generic applications (e.g., to support a risk-informed examination of proposed cable qualification standards).

- The workshop participants, mirroring the broader PRA community, have differing point of views as to whether the results of Fire PRA (and PRA in general) should be used to demonstrate that a plant has achieved a prescribed safety level.

Methodological and technical challenges

As indicated above, work is ongoing to address a number of technical challenges in Fire PRA. Work on the following challenges is in a relatively early stage of development.

- **Multiple hazards and events:** The treatment of fires induced by other hazards and events (e.g., seismically-induced fires; fires resulting from turbine failures, possibly concurrent with explosions, turbine missiles, and even flooding; fires resulting from operator actions responding to an initiator) and of multiple hazards induced by fires (e.g., flooding from suppression efforts, explosions, secondary fires) poses significant phenomenological and combinatorial challenges. Systematic but efficient methods are needed to support rapid screening of the myriad possibilities, as well as to support a realistic analysis of potentially important contributors.

- ***Low power and shutdown operating states:*** Low power and shutdown Fire PRAs have been performed (e.g., [49]) and guidance is available (e.g. [40]). However, the methods available to date are generally adaptations of at-power methods. Aside from adjustment issues (e.g., how to estimate fire ignition frequencies and the likelihoods of fire barrier failures given the different plant conditions during shutdown), methodological challenges include HRA (e.g., how to model and quantify human actions during shutdown conditions, let alone during shutdown fires) and the treatment of extended duration accidents (for which a realistic analysis might need to deal with offsite resources). These analysis challenges may increase when dealing with decommissioned plants, which will require the treatment of spent fuel accidents, and which could benefit from systematic and efficient screening approaches.

- ***Multi-unit impacts:*** Although not a major topic of discussion at the workshop, it is widely recognised that, in light of previous fire-related “near misses” (starting with the 1975 Browns Ferry fire[5]) as well as the broader lessons from the Fukushima Dai-ichi reactor accidents, the fire risk associated with multi-unit events is worth examining. The methods of multi-compartment analysis are likely applicable to such an examination, but a broader, more systematic analysis

[5] Other notable fire events involving multiple units include Greifswald (1975), Armenia (1982), and Narora (1993) [51].

(e.g., considering such interactions as shared water supplies) has yet to be pursued. The NRC's ongoing Level 3 PRA project [50], when completed, may shed some light on this topic.

4.1.2 Post-Fukushima Lessons

Although the Fukushima Dai-ichi reactor accidents were not "fire events" in the sense typically considered in Fire PRA (post-core damage explosions did affect the course of the accident), it is widely recognised that there are broader lessons from the accidents and subsequent stress tests that could be applicable to Fire PRA. Indeed, the Fire PRA workshop documented in this report was, in part, viewed by many WGRISK members as a natural post-Fukushima activity.

Three major points were raised during the workshop:

- There is a need to perform PRAs that address combinations of fires and other internal and external hazards (seismic, flooding, HEAF, explosions, etc.) in a systematic manner. One paper noted that over 10 % of the fire events in the OECD FIRE Database involved some form of combined effects. (Some of these events had a minor effect, while others led to the loss of a safety train.)

- A number of countries have upgraded their regulatory requirements to address combinations of events.

- It is recognised that some post-Fukushima changes aimed at certain scenarios (e.g., increases in fire water system inventory) can increase the risks associated with others (e.g., internal flooding).

4.1.3 Sharing of Methods, Good Practices, and Experiences

Overall, the workshop was successful in providing a venue for exchanging information on Fire PRA technology (methods, models, tools, and data) and applications. In addition to discussing the status of current activities, the participants were able to share their points of view on:

- key technical challenges (their nature, importance, and role in applications),

- possible technical solutions (including approaches being used in other fields that might be useful to the investigators),

- the accuracy and basis of stated results (e.g., utilisation of FDS uncertainty metrics in Fire PRA, empirical probabilities of open fire doors),

- the utility of Fire PRA (despite recognised weaknesses), and

- the need for improved co-operation between the operating experience and PRA communities (to ensure that operational experience databases and analyses address issues of high concern for PRAs, and to ensure that PRAs make best use of operational data).

The workshop and the subsequent writing session also provided an opportunity for participants to think about new WGRISK activities and products that could help PRA analysts and users.

4.2 Recommendations

Based on its review of the workshop conclusions and subsequent discussions, WGRISK has the following recommendations:

- Recognising that fire-related risk-informed applications are likely to remain important over the next few years, that the scope of such applications is increasing (e.g., to include Level 2 and low power and shutdown operations), and that Fire PRA research and applications are resulting in technological improvements, WGRISK should continue to remain active in the Fire PRA area. Potential (and non-exclusive) future activities include:
  - the organisation of a future workshop, perhaps focused on a specific aspect of Fire PRA, including:
    - multiple hazards/events (including fire),
    - fire HRA,
    - multi-unit Fire PRA,
  - participation in future conferences and other professional meetings;
  - development of additional resource or guidance documents (e.g., a SOAR, a condensed SOAR, a TOP, a focused guidance document/database based on member country experiences), as discussed in Section 3.11; and
  - participation (perhaps as a peer review organisation or as a source of peer reviewers) in international comparison and harmonisation activities (e.g., ASAMPSA_E).

Specific future activities will be pursued by WGRISK following its normal processes (i.e., developing a CSNI Activities Proposal Sheet, identifying a core group of participants, gaining working group approval, and gaining CSNI approval) [46].

Moreover, the workshop revealed recommendations being not specific for risk analysis only, however being an outcome of general discussions held during the workshop and naturally incorporating it. These recommendations listed in the following may be benefitial for CSNI and CNRA in general.

- Recognising the value of risk information in focusing operational experience data collection and analysis, and the value of operational experience data in shaping the development of PRA models and in the quantification of PRA model parameters, WGRISK should:
  - strengthen its ties with international operational experience working groups particularly the NEA Committee of Nuclear Regulatory Authorities Working Group on Operating Experience (WGOE); and
  - support efforts to more broadly disseminate risk-relevant lessons from operating experience reviews (e.g., such as those discussed at the annual Technical Meeting on Experiences with Risk-based Precursor Analysis in Belgium, whose conduct and findings may not be widely known within the PSA community).

- As indicated in Section 1 of this report, WGRISK has carried out numerous activities in Fire PRA. However, it has not tracked the outcomes and recommendations (detailed as well as general) resulting from these activities. Therefore it is recommended that WGRISK should develop and implement a process to systematically record and track activity conclusions and recommendations, in order to ensure the full knowledge of past efforts is built into future activities.

- Finally, although the role of WGRISK is to share information and experience related to risk assessment it is recognised that this work can also be potentially useful direct input to nuclear safety and the development of safety standards used by member organisations. Thus, WGRISK should strengthen its links with standards development organisations including IAEA (which participates in WGRISK meetings). This would likely produce mutual benefit to WGRISK (both in ensuring use of its products and in identifying areas of need) and these other organisations (e.g. by providing information supporting improved standards and guidance).

5. REFERENCES

[1] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), *Proceedings of OECD/NEA Workshop on Fire Risk*, 26 June-2 July 1999, Helsinki, Finland, NEA/CSNI/R(99)26, June 2000, www.oecd-nea.org/nsd/docs/1999/csni-r99-26.pdf.

[2] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), *Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics: State-of-the-art Report*, NEA/CSNI/R(99)27, March 2000, http://search.oecd.org/officialdocuments/displaydocumentpdf/?cote=NEA/CSNI/R(99)27&docLanguage=En.

[3] 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 (2002-2008) – First Applications and Expected Further Developments'*, NEA/CSNI/R(2009)6, Paris, September 2009, www.oecd-nea.org/nsd/docs/2009/csni-r2009-6.pdf.

[4] Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI), PRISME experimental project from 2006 to June 2012 - Application of the OECD PRISME results to investigate heat and smoke propagation mechanism in multi-compartment fire scenarios, NEA/CSNI/R(2012)14.

[5] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), *The Use and Development of Probabilistic Safety Assessment in NEA Member Countries*, NEA/CSNI/R(2002)18, Paris, France, July 2002, www.oecd-nea.org/nsd/docs/2002/csni-r2002-18.pdf.

[6] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), *Use and Development of Probabilistic Safety Assessment*, NEA/CSNI/R(2007)12, Paris, France, November 2007, www.oecd-nea.org/nsd/docs/2007/csni-r2007-12.pdf.

[7] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), *Use and Development of Probabilistic Safety Assessment: An Overview of the Situation at the End of 2010*, NEA/CSNI/R(2012)11, Paris, France, January 2013, http://search.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/CSNI/R(2012)11&docLanguage=En.

[8] International Atomic Energy Agency (IAEA), *Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants*, SSG-3, IAEA Safety Standards Series No. SSG-3, STI/PUB/1430, ISBN 978-92-0-114509-3, Vienna, Austria, April 2010, www-pub.iaea.org/MTCD/publications/PDF/Pub1430_web.pdf.

[9] Western European Nuclear Regulators' Association (WENRA) Reactor Harmonisation Working Group (RHWG): *WENRA Reactor Safety Reference Levels*, January 2008, www.wenra.org/media/filer_public/2012/11/05/list_of_reference_levels_january_2008.pdf.

[10] Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), International Workshop on Fire PSA, Puerto Vallarta, Mexico, 23-26 May 2005, http://home.nea.fr/download/wgrisk/Workshoppapers.zip.

[11] Kassawara, R. P., J. S. Hyslop, et al., *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 1: Summary & Overview*, EPRI TR 1011989, Electric Power Research Institute (EPRI), Palo Alto, CA 94303, USA / NUREG/CR-6850, Final Report, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research (RES), U.S. Nuclear Regulatory Commission (NRC), Rockville, MD 20852-2738, USA, September 2005, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v1/cr6850v1.pdf.

[12] Kassawara, R. P., J. S. Hyslop, et al., *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*, *Volume 2: Detailed Methodology*, EPRI TR 1011989, Electric Power Research Institute (EPRI), Palo Alto, CA 94303, USA / NUREG/CR-6850, Final Report, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research (RES), U.S. Nuclear Regulatory Commission (NRC), Rockville, MD 20852-2738, USA, September 2005, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/v2/-

[13] Kassawara, R. P., J. S. Hyslop, et al., *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*, *Methods Enhancements*, EPRI TR 1019259, Electric Power Research Institute (EPRI), Palo Alto, CA 94303, USA / NUREG/CR-6850, Supplement 1, Technical Report, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research (RES), U.S. Nuclear Regulatory Commission (NRC), Rockville, MD 20852-2738, USA, September 2010, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/s1/cr6850s1.pdf.

[14] The American Society of Mechanical Engineers (ASME), *Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa–2009*, *2009*, www.ewp.rpi.edu/hartford/~povron/EP/Other/ASME-ANS%20RA-Sa-2009.pdf.#

[15] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), *Verification and Validation of Selected Fire Models for NPP Applications*, *Volumes 1-7,* NUREG-1824, Washington, DC 20555-0001, USA, / EPRI 1011999, Electric Power Research Institute (EPRI), Palo Alto, CA 94303, USA, 2007 www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1824/.

[16] Salley, M. H., R. Wachowiak, et al., *Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG) — Final Report*, NUREG-1934, U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA, / EPRI u1023259, Electric Power Research Institute (EPRI), Palo Alto, CA 94303-1338, USA, November 2012, www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1934/.

[17] Nowlen, S. P., et al., *A Framework for Low Power/Shutdown Fire PRA – Final Report*, NUREG/CR-7114, U.S. Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA / SAND2011-0027P, Sandia National Laboratories (SNL), Risk and Reliability Analysis Department, Albuquerque, NM 87185, USA, September 2013, http://pbadupws.nrc.gov/docs/ML1326/ML13260A155.pdf.

[18] Cooper, S., *EPRI/NRC-RES Fire Human Reliability Analysis Guidelines*, EPRI TR 1019196, Electric Power Research Institute (EPRI), Palo Alto, CA 94303, USA, NUREG-1921, Final Report, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research (RES), U.S. Nuclear Regulatory Commission (NRC), Washington, DC 20555-0001, USA, Technical Update November 2009, http://pbadupws.nrc.gov/docs/ML0933/ML093350494.pdf.

[19] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke, *Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke* (*Methods for Probabilistic Safety Analysis of Nuclear Power Plants*), Stand: August 2005, BfS-SCHR-37/05, Salzgitter, Germany, October 2005 (in German).

[20] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke, *Daten zur Quantifisierung von Ereignisablaufdiagrammen und Fehlerbäumen* (*Data for Quantification of Event Sequence Diagrams and Fault Trees*), Stand: August 2005, BfS-SCHR-38/05, October 2005 (in German).

[21] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke, *Ergänzungen zu Methoden und Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke* (*Addenda on PSA Methods and Data for Nuclear Power Plants*), Draft, 06.10.2014.

[22] Nowlen, S. P.,F. J. Wyant, K. McGrattan, et al., *Cable Response to Live Fire (CAROLFIRE), Volumes 1-3*, NUREG/CR-6931, U.S. Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA / SAND2007-600, Sandia National Laboratories (SNL), Risk and Reliability Analysis Department, Albuquerque, NM 87185, USA, April 2008, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6931/.

[23] McGrattan, K., et al., *Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE*), *Volumes 1-2*, NUREG/CR-7010, U.S. Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA, July 2012, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7010/.

[24] Nowlen, S. P., J. W. Brown, et al., *Kerite Analysis in Thermal Environment of FIRE (KATE-Fire): Test Results – Final Report*, NUREG/CR-7102, U.S. Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA / SAND2011-6548P, Sandia National Laboratories (SNL), Risk and Reliability Analysis Department, Albuquerque, NM 87185, USA, December 2011, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7102/.

[25] Salley, M. H., R. Wachowiak, et al., *Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE): Final Report, Volume 1*, NUREG-7150, BNL-NUREG-98204-2012, U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA, / EPRI 1026424, Electric Power Research Institute (EPRI), Palo Alto, CA 94303-1338, USA, October 2012, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7150/v1/.

[26] Subudhi, M., G. Taylor, R. Wachowiak, et al., *Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE): Final Report, Volume 2*, NUREG-7150, BNL-NUREG-98204-2012, U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES), Washington, DC 20555-0001, USA, / EPRI 3002001989, Electric Power Research Institute (EPRI), Palo Alto, CA 94303-1338, USA, May 2014, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7150/v2/.

[27] Audoin, L. et al., "*Quantifying differences between computational results and measurements in the case of large scale well-confined fire scenarios*", Nuclear Engineering and Design 241 (1). Elsevier; pp. 18-31, January 2011.

[28] Audoin, L., H. Prétrel, W. Le Saux, "*Overview of the OECD PRISME Project – Main Experimental Results*", in: Proceedings of SMiRT 21, 12th International Seminar on Fire Safety in Nuclear Power Plants and Installations. München, Germany, September 13-15, 2011, GRS-A-3651, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany, September 2011.

[29] OECD / Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) Working Group IAGE (WGIAGE), *NEA CSNI WGIAGE TASK ON HIGH ENERGY ARCING FAULT EVENTS (HEAF),* NEA/CSNI/R(2015)10, Paris, 2015.

[30] OECD / Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) Working Group IAGE (WGIAGE), *OECD High Energy Arcing Fault Events Joint Nuclear Safety Research Project*, www.oecd-nea.org/jointproj/heaf.html

[31] Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI), *OECD PRISME-2 Joint Nuclear Safety Research Project*, www.oecd-nea.org/jointproj/prisme-2.html

[32] Röwekamp, M., J. Dreisbach, S. Miles, et al., *International Collaborative Fire Modeling Project (ICFMP) - Summary of Benchmark Exercises 1 to 5*, GRS-227, Gesellschaft für Anlagen- und

Reaktorsicherheit (GRS) mbH, ISBN-Nr.: 978-3-939355-01-4, Köln, Germany, September 2008, www.grs.de/content/grs-227-international-collaborative-fire-modeling-project-icfmp.

[33] Hostikka, S., O. Keski-Rahkonen, *Probabilistic simulation of fire scenarios*, Nuclear Engineering and Design 224(3), pp. 301-311, 2003.

[34] Electric Power Research Institute (EPRI): *The Updated Fire Events Database: Description of Content and Fire Event Classification Guidance,* EPRI-1025284, Palo Alto, CA, USA, July 2013.

[35] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), *Use of OECD Data Project Products in Probabilistic Safety Assessment*, NEA/CSNI/R(2014)2, Paris, June 2014, http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/CSNI/R(2014)2&docLanguage=En.

[36] Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI), *OECD FIRE Database*, *OECD FIRE DB 2013:1*, Paris, France, August 2014.

[37] National Fire Protection Association (NFPA), *Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants*, NFPA 805, 2010.

[38] United States Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research, Severe Accidents Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, Washington, DC, December 1990, www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/.

[39] Berry, D. L, E. E. Minor, *Nuclear Power Plant Fire Protection - Fire-Hazards Analysis (Subsystems Study Task* 4), prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-0654 and SAND79-0324, September 1979, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr0654/.

[40] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), “Safety Requirements for Nuclear Power Plants”, Federal Gazette, January 24, 2013 (in German, to be published in English), www.bfs.de/de/bfs/recht/rsh/volltext/3_BMU/3_0_1_1112.pdf.

[41] Siu N. O., *Modeling issues in nuclear plant fire risk analysis*, Proceedings of EPRI Workshop on Fire Protection in Nuclear Power Plants, EPRI NP-6476, August 1989.

[42] Chavez, J. M., S. P. Nowlen, *An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets*, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4527 and SAND86-0336, Washington, DC, November 1988, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr4527/.

[43] Budnitz, R. J., et al., *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts*, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-6372, Vols. 1 and 2, 1997, www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6372/.

[44] International Atomic Energy Agency (IAEA), Determining the quality of probabilistic safety assessment (PSA) for applications in nuclear power plants, IAEA.TECDOC 1511, Vienna, Austria, July, 2006, www-pub.iaea.org/MTCD/publications/PDF/te_1511_web.pdf.

[45] Nuclear Energy Institute (NEI), *Guidance for Post-Fire Safe-Shutdown Circuit Analysis*, NEI-00-01, Revision 3, October 2011, http://pbadupws.nrc.gov/docs/ML1129/ML112910147.pdf.

[46} Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), Working Group on Risk (WGRISK), *OECD WGRISK Integrated Plan*, www.oecd-nea.org/nsd/csni/wgrisk.html.

[47} Siu, N. O., N. Melly, S. P. Nowlen, M. Kazarians, *Fire Risk Assessment for Nuclear Power* Plants, The SFPE Handbook of Fire Protection Engineering, 5th Edition, National Fire Protection Association /NFPA), Quincy, MA, in publication (ADAMS ML XXXXXXXXX).

[48] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI*), CSNI Technical Opinion Paper No. 1: "Fire Probabilistic Safety Assessment for Nuclear Power Plants*, Paris, 2002.

[49] Musicki, Z., et al., *Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1: Analysis of Core Damage Frequency from Internal Fires During Mid-Loop Operations*, NUREG/CR-6144, Vol. 3, 1994.

[50] U.S. Nuclear Regulatory Commission, *Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities*, SECY-11-0089, (ADAMS ML11090A039), Washington, DC, 2011.

[51] Siu, N. O., M. Stutzke, *Technical Challenges in Multi-Unit Fire PSA*, in: Proceedings of International Workshop on Multi-Unit Probabilistic Safety Assessment (PSA), Ottawa, ONT, Canada, November 17-20, 2014 (in preparation) (ADAMS ML14282A263).

[52] 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.

APPENDIX 1: CSNI ACTIVITY PROPOSAL SHEET (CAPS) WGRISK (2012)-2, "INTERNATIONAL WORKSHOP ON FIRE PRA"

<table>
<tr><th>Project/Activity Title</th><th>Workshop on FIRE PRA in Member Countries</th></tr>
<tr><td>Objectives</td><td>Organise a workshop on the state-of-the-art methods for quantitative fire risk assessment of NPPs and associated applications in member countries.

One objective of the workshop is to develop recommendations regarding a potential future update of the State-of-the-art Report on fire risk analysis (NEA/CSNI/R(99)27 including further development of methods for fire risk analysis, collection of operating experience and processing of data to be used in Fire PRA applications.</td></tr>
<tr><td>Scope/Justification/

Deliverables, Expected results and users, Relation to other projects</td><td>Scope

The task comprises a review of current experiences in using methods for fire risk analysis in modelling and quantification of accident sequences induced by fires.

General topics to be addressed include:

• Development of joint probabilistic-deterministic fire spreading scenarios,
• Fire simulation applications,
• Defence-in-depth in fire protection measures, particularly considering large fire loads,
• Modeling of manual fire fighting capabilities (administrative fire protection).

Specific topics to be addressed are:

• Determination of fire frequencies for rooms and components,
• Fire experiments,
• Fire models of solid materials,
• Modeling of fire spreading on cables,

Human and organisational factors.

Justification

Since the publication of OECD/CSNI/PWG5 on the State-of-the-art Report on fire (NEA/CSNI/R(99)27 "Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics", February 2000) the development of fire risk assessment has made major steps. The follow-up workshop in Mexico in 2005 indicated that important new developments had taken place and also addressed needs for further development.

Examples of developments include:

• USA: Fire PRA Methodology for Nuclear Power Facilities (EPRI 1011989 / NUREG/CR-6850)</td></tr>
</table>

	• USA: Fire Probabilistic Risk Assessment Methods Enhancements (EPRI 1019259/ NUREG/CR-1921 (NURE/CR-6850 Supplement 1) • USA: Fire PRA Methodology (ANSI/ANS-58.23-2007) • USA: Verification and Validation of Selected Fire Models for NPP Applications (EPRI 1011999 / NUREG-1824) • USA: Fire Modeling Application Guide (EPRI 1023259 / NUREG-1934 Draft) • USA: Methodology for Low Power/Shutdown Fire PRA (NUREG/CR-7114 Draft) • USA: Fire Human Reliability Analysis Guidelines (EPRI 1019196 / NUREG/CR-1921 Draft) • USA: Research projects on cable fires (e.g. CAROLFIRE; CHRISTFIRE; KATE-Fire) • Germany: (PSA Methods and Data, Reports BfS-SCHR-37/05 and -38/05,in German) • Germany: Methods for Conducting Fire PSA for Low power and Shutdown States, GRS Report, GRS-A-3579 • France: Development of fire simulation code SYLVIA code based on an experimental fire research programme Finland: Development of joint probabilistic-deterministic fire spreading scenarios using FDS code (applications for I&C cabinet rooms, cable tunnels and cable spreading rooms); fire research of flame retardant non corrosive cables. **Deliverables:** Workshop proceedings. CSNI/WGRISK is proposing a workshop on fire risk analysis in 2013. The workshop programme is to be planned in such a way that the proceedings will help in a potential further step to update the State-of-the-art Report on fire risk analysis (NEA/CSNI/R(99)27. **Expected users:** • Regulatory organisations • Risk analysis practitioners • Individuals responsible for plant safety management **Relation to other projects:** • Fire Incident Records Exchange Project (OECD FIRE) • OECD PRISME and PRISME2 projects • National fire research projects in member countries

Safety significance/ priority (see priority criteria in Section IV)	Regarding the priority criteria set in Section IV of the CSNI Operating Plan: • Criterion 1: Relevance to CSNI challenges and technical goals. • Criterion 2: Better accomplished by international group. • Criterion 3: Likely to bring conclusive results in reasonable time frame.
Technical Goal(s) covered	The following CSNI Main Challenges (and associated Technical Goals) are addressed by the proposed project: 3g) To further review and assess the development of PSA methods; 3h) To review and assess safety approaches related to radioactivity confinement, criticality, fire and chemical risks in nuclear installations; 3i) To contribute to the enhancement of safety performance of current nuclear installations
Knowledge Management and Transfer covered	Results of this activity will be documented in a CSNI report to facilitate knowledge transfer.
Milestones (deliverables vs. time)	• Planning and nominations for task group (Summer 2012) • First meeting, detailed work plan (Fall 2012) • Second meeting (Spring 2013) • Workshop (Summer / early fall 2013) • Report to the CSNI (June 2014)
Lead organisation(s) and co-ordination	Lead: GRS, Germany Countries expressing interest to be active participants include Canada, Chinese Taipei, Finland, France, Germany, India, Mexico, Slovak Republic, Spain, UK and USA.
Participants (individuals and organisations)	Experts from countries with known experience in the topic will be invited to the planning meeting and to contribute to active project work. Representatives from all WGRISK member countries are invited to take part in the later Working Meetings. Participation of countries having Fire PRA experience will be strongly encouraged to participate in the workshop.
Resources	For the core task group organisations the effort will consist of planning and organising the workshop and preparing the workshop proceedings. The overall effort is estimated to amount to approximately 1-2 man-years. For each other participant, attending the workshop and review of the proceedings will take approximately 2 man-weeks.
Action from CSNI	Approved

APPENDIX 2: WORKSHOP AGENDA

Monday, 28 April 2014

08:00 h	**Registration**	
	Opening Session	***Chair:*** ***Marina Röwekamp (GRS, Germany)***
10:00 h	Welcome address	*Andreas Schaffrath (GRS, Germany)*
10:10 h	Introduction and Opening Remarks	*Marina Röwekamp (GRS, Germany)*
10:15 h	Keynote Speech	*Nathan Siu (NRC, USA)*
10:35 h	NEA Nuclear Safety Division – The Structures and Processes that Support Projects and Working Groups to Deliver Good Practice Guidance	*Neil Blundell (NEA, France)*
11:00 h	***Coffee Break***	
	Session 1: Progress in Fire PRA methodology - relevant issues and methods	***Chairperson:*** ***Per Hellström (SSM, Sweden)***
11:20 h	Fire PRA Maturity and Realism: A Technical Evaluation and Questions	*Nathan Siu (NRC, USA)*
11:45 h	Fire Ignition Frequency Estimation: Component-based versus Room-based Approaches	*Roman Grygoruk (AREVA NP, Germany)*
12:10 h	Probabilistic Set of Filter Criteria in the Frame of Fire PSA	*Florian Berchtold (BAM, Germany)*
12:35 h	***Lunch Break***	
	Session 2: Progress in Fire PRA methodology - tools and data	***Chairperson:*** ***Marina Röwekamp (GRS, Germany)***
14:00 h	Fire PSA Automatisation Tool	*Albert Malkhasyan (TRACTEBEL, Belgium)*
14:25 h	Simulation Toolbox Development for Fire PRA	*Simo Hostikka (Aalto University, Finland)*
14:50 h	Building up A Cable Management System and its Use for Fire PRA	*Andreas Hempel (KABTec, Germany)*
15:20 h	***Coffee Break***	
	Session 2: Progress in Fire PRA Methodology - Tools and Data (contd.)	***Chairperson:*** ***Marina Röwekamp (GRS, Germany)***
15:40 h	Failure Mode and Effect Analysis of Cable Failures for Fire PSA with Quantification of Failure Mode Probabilities	*Joachim Herb (GRS, Germany)*
16:05 h	Extension of the German Database of Plant Specific Failure Rates for Fire Protection Systems and Components	*Burkhard Forell (GRS, Germany)*
16:30 h	***Coffee Break***	
	Session 3: Use and application of fire event databases	***Chairperson:*** ***Nicholas Melly (NRC, USA)***
16:45 h	Fire Events Database – Insights and Opportunities from the Fire Event Database	*Ashley Lindeman (EPRI, USA)*

17:10 h	Use and Applicability of the OECD FIRE Database Project	*Marina Röwekamp, (GRS, Germany)*
17:35 h	***Adjourn of the first day***	
17:45 h	***Welcome Reception***	***all participants***

Tuesday, 29 April 2014

	Session 3: Use and Application of Fire Event Databases (contd.)	***Chairperson:*** ***Nicholas Melly (NRC, USA)***
09:15 h	First Investigations Regarding Combinations of Fires and other Events in the OECD FIRE Database	*Heinz-Peter Berg (BfS, Germany)*
09:45 h	***Coffee Break***	
	Session 4: Fire Safety Analysis and Research	***Chairperson:*** ***Simo Hostikka (Aalto University, Finland)***
10:00 h	U.S. NRC Fire Safety Research Activities	*Nicholas Melly (NRC, USA)*
10:25 h	Fundamental Investigation on Successive Fire Due to the High Energy Arcing Faults Event for the High Voltage Switchgears	*Koji Shirai (CRIEPI, Japan)*
10:50 h	Expert Judgement: An Application in Fire Induced Circuit Analysis	*Nicholas Melly (NRC, USA)*
11:20 h	***Coffee Break***	
11:35 h	Predicting the Heat Release Rate of Liquid Pool Fires Using CFD	*Topi Sikanen (VTT, Finland)*
12:00 h	Predicting Fire Spread of A Cable Using Methods of Pyrolysis Modeling	*Anna Matala (VTT, Finland)*
12:30 h	***Lunch Break***	
	Session 5: Requirements and Guidance for Fire PSA	***Chairperson:*** ***Heinz-Peter Berg (BfS, Germany)***
14:00 h	Insights on Recent Regulatory Guideline of Fire Protection in Finland	*Matti Lehto (STUK, Finland)*
14:25 h	Regulatory Framework and Insights from Fire PSA of Canadian Nuclear Power Plants	*Usha Menon (CNSC, Canada)*
14:50 h	Guidance for Performing Fire PRA in Germany	*Heinz-Peter Berg (BfS, Germany)*
15:15 h	Fire PSA Attributes in the Integrated IAEA Guidelines for PSA Quality	*Artur Lyubarskiy (IAEA, Austria)*
15:45 h	***Coffee Break***	
	Session 6: Insights from Probabilistic Fire Analyses	***Chairperson:*** ***Matti Lehto (STUK, Finland)***
16:15 h	Focus on the Studies in Support of Fire PSA of the French 1300 MWe Nuclear Power Plants	*Julien Espargilliere (IRSN, France)*
16:40 h	Insights from HRA and MSOs Analyses in Spanish Fire PRA	*Pedro Fernández Ramos (Empresarios Agrupados, Spain)*
17:10 h	***Adjourn of the second day***	
18:00 h	***Hosted Dinner Event (organised with bus transfer from Garching)***	***all participants***

Wednesday, 30 April 2014

	Session 6: Insights from Probabilistic Fire Analyses (contd.)	***Chairperson:*** ***Matti Lehto (STUK, Finland)***
09:15 h	Uses of Fire PSA and Sensitivity Studies	*Fabienne Nicoleau (IRSN, France)*
09:40 h	Fire PRA and Multiple Spurious Operations Study in Korea UCN 3 Unit	*Kilyooo Kim (KAERI, Korea)*
10:05 h	Spurious Operation Scenarios - Knowledge Sharing from the MSO Mapping of Ringhals PWRs	*Anna Bjereld (Ringhals AB, Vattenfall, Sweden)*
10:35 h	***Coffee Break***	
	Session 7: Fire PSA Applications	***Chairperson:*** ***Rémy Bertrand (IRSN, France)***
10:55 h	The Fire on Turbine Generator as an Important Fire Risk Contributor at NPP Dukovany (VVER-440)	*Ladislav Kolar (UJV Rez, Czech Republic)*
11:20 h	Updating of the Fire PRA of the Olkiluoto NPP Units 1 and 2	*Lasse Tunturivuori (TVO, Finland)*
11:45 h	Experience from Developing and Implementing Shutdown Fire PRA at Forsmark NPP	*Erik Cederhorn (Riskpilot AB, Sweden)*
12:15 h	***Lunch Break***	
	Session 8: Extension of Fire PRA Scope and Applications	***Chairperson:*** ***Rémy Bertrand (IRSN, France)***
13:45 h	Latest Extensions of the Loviisa Fire PRA	*Antti Paajanen (FORTUM, Finland)*
14:10 h	Conducting Fire PSA for the Post-commercial Shutdown Phase	*Michael Türschmann (GRS, Germany)*
14:40 h	***Coffee Break***	
	Concluding Session	***Chairperson:*** ***Marina Röwekamp (GRS, Germany)***
15:00 h	Workshop Summary, Final Discussion and Concluding Remarks	*all session chairpersons*
16:00 h	**Workshop Adjourn**	

APPENDIX 3: LIST OF WORKSHOP PARTICIPANTS

BELGIUM

Federico Agosti
TRACTEBEL SA – GDF SUEZ
Avenue Ariane 7
1200 Bruxelles
Telephone: +32 (0) 2 773 7493
E-mail: federico.agosti@gdfsuez.com

Frederick Bonte
Bel V
Rue Walcourt 148
1070 Brussels
Telephone: +32 (0) 2 528 0254
E-mail: Frederick.Bonte@belv.be

Philippe Dejardin
TRACTEBEL SA – GDF SUEZ
Avenue Ariane 7
1200 Bruxelles
Telephone: +32 (0) 2 773 8558
E-mail: philippe.dejardin@gdfsuez.com

Veronique Jacques
Bel V
Rue Walcourt 148
1070 Brussels
Telephone: +32 (0) 2 528 0265
E-mail: Veronique.Jacques@belv.be

Albert Malkahasyan
TRACTEBEL SA – GDF SUEZ
Avenue Ariane 7
1200 Bruxelles
Telephone: +32 (0) 2 773 7016
E-mail: albert.malkhasyan@gdfsuez.com

CANADA

Usha Menon
Canadian Nuclear Safety Commission (CNSC)
280 Slater St.
Ottawa, ON, K1P 5S9
Telephone: +1 613 947 8222
E-mail: Usha.Menon@cnsc-ccsn.gc.ca

CZECH REPUBLIC

Ladislav Kolar
UJV Rez, a. s.
Hlavni 130
Husinec Rez, 250 68
Telephone: 42 (0) 266 173 561
E-mail: kol@ujv.cz

FINLAND

Simo Hostikka
Aalto University School of Engineering
P.O. Box 12100
00076 Aalto
Telephone: +358 50 447 1582
E-mail: simo.hostikka@aalto.fi

Matti Lehto
Radiation and Nuclear Safety Authority – STUK
P.O. Box 14
00881 Helsinki
Telephone: +358 40 0888 167
E-mail: matti.lehto@stuk.fi

Anna Matala
VTT Technical Research Centre of Finland
P.O. Box 1000
02044 VTT
Telephone: +358 20 722 4207
E-mail: Anna.Matala@vtt.fi

Antti Paajanen
FORTUM Power and Heat Oy
Keilaniemi, POB 100
00048 FORTUM
Telephone: +358 40 35 86570
E-mail: antti.paajanen@fortum.com

Topi Sikanen
VTT Technical Research Centre of Finland
P.O. Box 1000
02044 VTT
Telephone: +358 20 722 4340
E-mail: Topi.Sikanen@vtt.fi

Lasse Tunturivuori
Teollisuuden Voima OYJ
Olkiluoto
27160 EURAJOKI
Telephone: +358 (02) 83811
E-mail: Lasse.Tunturivuori@tvo.fi

FRANCE

Laurent Audouin
Institut de Radioprotection et de Sûreté Nucléaire
(IRSN) PSN-RES/SA2I/LEF
CE de Cadarache - Bât 346 - BP 3
13115 - Saint Paul Lez Durance Cedex
Telephone: +33 (0) 4 42199408
E-mail: laurent.audouin@irsn.fr

Remy Bertrand
Institut de Radioprotection et de Sûreté Nucléaire
(IRSN)
31 avenue de la Division Leclerc B.P. 17
92262 Fontenay-aux-Roses CEDEX
Telephone: + 33 (0) 1 58359105
E-mail: remy.bertrand@irsn.fr

Julien Espargilliere
Institut de Radioprotection et de Sûreté Nucléaire
(IRSN)
31 avenue de la Division Leclerc B.P. 17
92262 Fontenay-aux-Roses CEDEX
Telephone: +33 (0) 1 58358857
E-mail: julien.espargilliere@irsn.fr

Gabriel Georgescu
Institut de Radioprotection et de Sûreté Nucléaire
(IRSN)
31 avenue de la Division Leclerc B.P. 17
F-92262 Fontenay-aux-Roses CEDEX
Telephone: +33 (0) 1 58358108
E-mail: gabriel.georgescu@irsn.fr

Bruno Laurent
Institut de Radioprotection et de Sûreté Nucléaire (IRSN)
31 avenue de la Division Leclerc B.P. 17
F-92262 Fontenay-aux-Roses CEDEX
Telephone: +33 (0)1 58358368
E-mail: bruno.laurent@irsn.fr

Fabienne Nicoleau
Institut de Radioprotection et de Sûreté Nucléaire (IRSN)
31 avenue de la Division Leclerc B.P. 17
92262 Fontenay-aux-Roses CEDEX
Telephone: +33 (0) 1 58357726
E-mail: fabienne.nicoleau@irsn.fr

GERMANY

Dagmar Baumann
TÜV Süd Energietechnik GmbH
Dudenstr. 28
68167 Mannheim
Telephone: +49 (0) 621 395542
E-mail: dagmar.baumann@tuev-sued.de

Florian Berchtold
BAM Federal Institute for Materials Research and Testing
Unter den Eichen 87
12205 Berlin
Telephone: +49 (0) 30 8104 4022
E-mail: florian.berchtold@bam.de

Heinz-Peter Berg
Bundesamt für Strahlenschutz (BfS), SK
Willy-Brandt-Str. 5
38226 Salzgitter
Telephone: +49 (0) 30 18333 1501
E-mail: hberg@bfs.de

Martin Boesel
Siemens AG
Freyeslebenstr. 1
91058 Erlangen
Telephone: +49 (0) 9131 18 4668
E-mail: martin.boesel@siemens.com

Burkhard Forell
Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Schwertnergasse 1
50667 Köln
Telephone: +49 (0) 221 2068 625
E-mail: Burkhard.Forell@grs.de

Roman Grygoruk
AREVA GmbH
Paul-Gossen-Str. 100
91052 Erlangen
Telephone: +49 (0) 9131 90093206
E-mail: roman.grygoruk@areva.com

Andreas Hempel
KAB-tec GmbH
Lyoner Str. 44-48
60528 Frankfurt a. M.
Telephone: +49 (0) 69 669099 25
E-mail: HempelA@kab-tec.de

Matias Krauß
Bundesamt für Strahlenschutz (BfS), SK
Willy-Brandt-Str. 5
38226 Salzgitter
Telephone: +49 (0) 30 18333 1540
E-mail: mkrauss@bfs.de

Christian Neugebauer
E.ON Kernkraft GmbH
Tresckowstr. 5
30457 Hannover
Telephone: +49 (0) 511 4393373
E-mail: christian.neugebauer@eon.com

Ewgenij Piljugin
Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Forschungszentrum, Boltzmannstr. 14
85748 Garching
Telephone: + 49 (0) 89 32004 470
E-mail: Ewgenij.Piljugin@grs.de

Marina Röwekamp
Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Schwertnergasse 1
50667 Köln
Telephone: +49 (0) 221 2068 898
E-mail: Marina.Roewekamp@grs.de

Michael Schwarz
EnBW Kernkraft GmbH
Rheinschanzinsel
D-7661 Philippsburg
Telephone: +49 (0) 7256 95 12131
E-mail: mi.schwarz@kk.enbw.com

Michael Türschmann
Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Kurfürstendamm 200
10719 Berlin
Telephone: +49 (0) 30 88589 132
E-mail: Michael.Tuerschmann@grs.de

HUNGARY

Istvan Kelemen
CK –Trikolor Ltd.
Torok u. 2
1023 Budapest
Telephone: +36 1-315 1101
E-mail: kelemen.dnv@t-online.hu

Sandor Czako
CK –Trikolor Ltd.
Torok u. 2
1023 Budapest
Telephone +36 30 423 8934
E-mail: czako.dnv@t-online.hu

NETHERLANDS

J.M. Tenniglo
NRG Arnhem
Utrechtseweg 310, P.O.Box 9034
6800 ES Arnhem
Telephone: +31 6 100 64563
E-mail: tenniglo@nrg.eu
mailto:Hubert.Boxman@minvrom.nl

Hans Brinkman
NRG Arnhem
Utrechtseweg 310, P.O.Box 9034
6800 ESArnhem
Telephone: +31 (0)26 356 8553
E-mail: brinkman@nrg.eu

ITALY

Laura Frisoni
ENEL
Via Mantova 24
00198 Roma
Telephone: +39 (0) 683 059865
E-mail: laura.frisoni@enel.com

JAPAN

Yoshikane Hamagushi
Nuclear Regulation Authority, Japan (NRA)
1-9-9 Roppongi-First Building, Roppongi,
Minato-ku
Tokyo 106-8450
Telephone: +81 (3) 5114 2224
E-mail: yoshikane_hamagoshi@nsr.go.jp

Koji Shirai
Central Research Institute of Electric Power Industry
(CRIEPI)
1646 Abiko, Abiko-shi
Chiba-ken 270-1194
Telephone: +81 (70) 6568 9717
E-mail: shirai@criepi.denken.or.jp

KOREA (REPUBLIC OF)

Kilyoo Kim
Korea Atomic Energy Research Institute (KAERI)
Integrated Safety Assessment Division
P.O. Box 105, Yusung
Daejeon 305-600
Telephone: +82-1049347547
E-mail: kykim@kaeri.re.kr

SPAIN

Eunate Armananzas Albaisar
Consejo Sguridad Nuclear (CSN)
Pedro Justo Dorado Dellmans, 11
28040 Madrid
Telephone +34 913300871
E-Mail: eaa@csn.es

Pedro Fernandez Ramos
Empresarios Agrupados
Calle Magallanes 3
28015 Madrid
Telephone +34 913 098 000 7107
E-Mail: pfr@empre.es

Maria Teresa Vázquez
Nuclear Safety Council (CSN)
C/ Pedro Justo Dorado Delmans, 11
28040 Madrid
Telephone: +34 91 346 02 60
E-mail: tvm@csn.es

SWEDEN

Anna Bjereld
Ringhals AB, Vattenfall, RTSA
432 85 Väröbacka
Telephone: +46 (0) 340 668516
E-Mail: anna.bjereld@vattenfall.com

Erik Cederhorn
Risk Pilot AB
Parmmätargatan 7
11224 Stockholm
Telephone: +46 7-08570820
E-mail: erik.cederhorn@riskpilot.se

Torsten Leander
Forsmarks Kraftgrupp AB
74203 Östhammar
Telephone: +46 1 7382982
E-mail: let@forsmark.vattenfall.se

Per Hellström
SSM Swedish Radiation Safety Authority
Solna strandväg 96
17116 Stockholm
Telephone: +46 8 799 4038
E-mail: per.hellstrom@ssm.se

Jakob Christensen
Risk Pilot AB
Jungmansgatan 12
21119 Malmö
Telephone: +46 705 246672
E-Mail: jakob.christensen@riskpilot.se

SWITZERLAND

Dominik Hermann
Swiss Federal Nuclear Inspectorate, ENSI
Industriestr. 19
5200 Brugg
Telephone: +41 (0) 56 460 85 99
E-mail: dominik.hermann@ensi.ch

UNITED STATES OF AMERICA

Ashley Lindeman
Electric Power Research Institute (EPRI)
1300 West W.T. Blvd.
Charlotte, NC 28262
Telephone: +1 704 807 4102
alindeman@epri.com

Nicholas Melly
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Telephone: +1 301 251 7916
E-mail: Nicholas.Melly@nrc.gov

Nathan Siu
US Nuclear Regulatory Commission
MS CSB C07M
Washington, DC 20555
Telephone: +1 301 251 7583
E-mail: Nathan.Siu@nrc.gov

INTERNATIONAL ORGANISATIONS

Neil Blundell
NEA Nuclear Safety Division
Le Seine St-Germain
12 Boulevard des Iles
92130 Issy-les-Moulineaux
Telephone: +33 (0) 145 24 1052
E-mail: Neil.Blundell@oecd.org

Takayoshi Nezuka
NEA Nuclear Safety Division
Nuclear Safety Division
Le Seine St-Germain
12 Boulevard des Iles
92130 Issy-les-Moulineaux
Telephone: +33 (0) 145 24 1151
E-mail: Takayoshi.Nezuka@oecd.org

Artur Lyubarskiy
IAEA
Wagramer Str. 5
1400 Vienna, Austria
Telephone: +43 (0) 1 2600 22516
E-mail: A.Lyubarskiy@iaea.org

APPENDIX 4: WORKSHOP PAPERS

APPENDIX 4: WORKSHOP PAPERS

NEA Nuclear Safety Division – "The Structures and Processes that Support Projects and Working Groups to deliver Good Practice Guidance"

Neil Blundell
OECD NEA, Issy-Les Moulineaux, Paris, France

Abstract

This paper explains the structure of the NEA Secretariat, the roles of its standing committees and the Joint Nuclear Safety Research Projects and the efforts that are made to ensure that the member countries are brought together to cooperate in delivering underpinned best practice guidelines based on real data for PSA analysis.

The work on PSA within CSNI is highlighted and the links with the data being generated within both the experimental and the database joint projects are also highlighted.

The paper stresses the importance of member countries using the following to develop together best practice practical guidance in the field of nuclear safety and Fire PSA:

- *Forums for the exchange of information and the transfer of experience*
- *Cooperation between member countries*
- *International co-operative activities of interest to the NEA member countries*
- *Mutual cooperation with the other standing committees on matters of common interest*

The NEA secretariat is committed to assisting the member states deliver their on their objectives.

1. Introduction

NEA has had for over 30 years a mature responsive infrastructure between, now 31, member states in the field of nuclear safety. The overall structure of the NEA is shown in Figure 1 below.

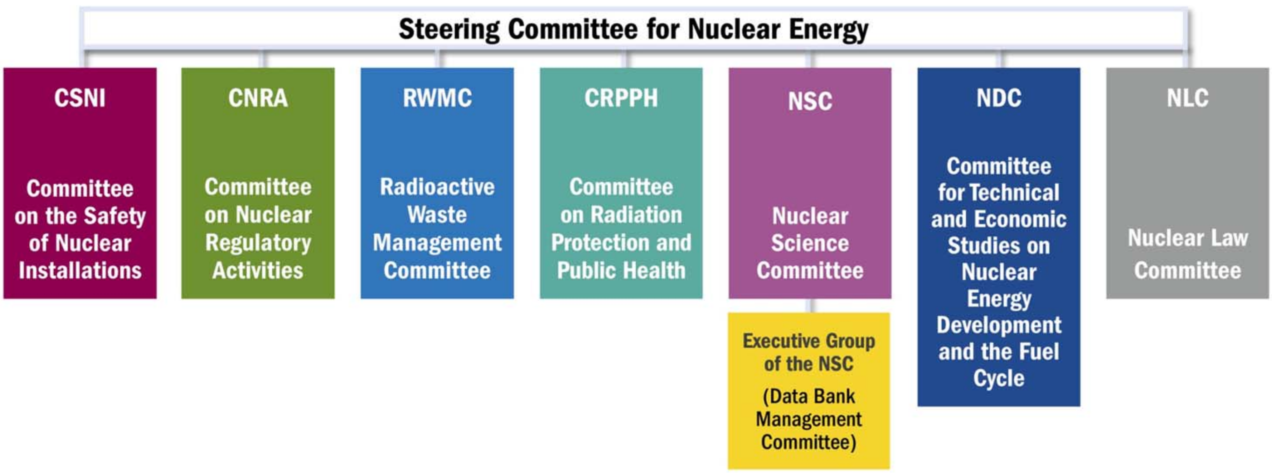


Figure 1. **Overall Committee Structure of NEA**

The NEA is a key international instrument of co-operation. It has eight standing Technical committees. The role of the standing technical committees is to develop this basic strength by fostering international co-operation in the NEA in their respective areas of work, advancing a common knowledge base and developing common approaches and consensus; also to optimise co-ordination among themselves and treat cross-cutting issues efficiently.

Three of the standing technical committees have safety mandates:

- CNRA (Committee on Nuclear Regulatory Activities)
- CSNI (Committee on the Safety of Nuclear Installations)
- CRPPH (Committee on Radiation protection and Public health)

In the area of safety there is also a large portfolio of Joint Nuclear safety research projects.

2. The Roles of the Standing Committees with Safety Mandates

Whilst the CRPPH is focused on radiological protection activities and CNRA on regulatory activities the mandates of the three committees have within them these common objectives related to safety matters:

- Provide a forum for the exchange of information and the transfer of experience
- Promote cooperation between member countries
- Promote and initiate international co-operative activities of interest to the NEA member countries
- Mutually cooperate with the other standing committees on matters of common interest.

An important task that CSNI has within its mandate is the objective of promoting the establishment of joint undertakings, and assisting in the feedback of the results to participating organisations. These undertakings are predominantly the joint nuclear safety research projects. As will be described later, these projects are initiated within CSNI.

In order to deliver their objectives the Standing Committees break their work down into specific workstreams managed by working groups (or parties) who in turn, where appropriate, set up task groups to deliver specific tasks.

The general structure and reporting relationships are shown in Figure 2.

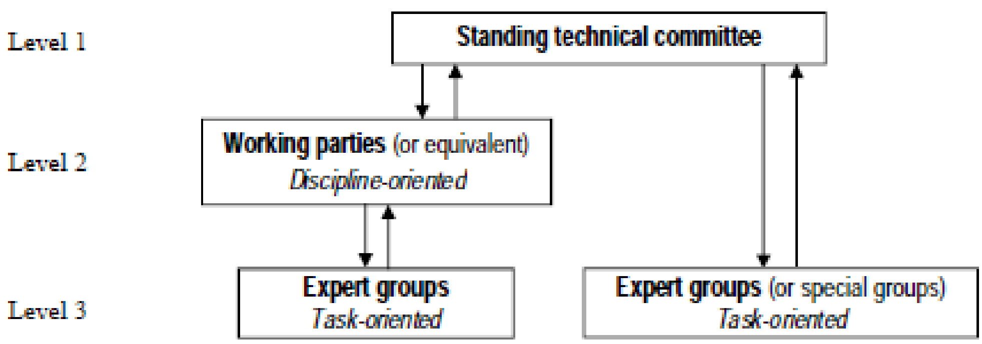


Figure 2. **Structure and Reporting within Standing Technical Committees**

It is important to recognise that the duration of the mandates has been set at:

- six years for level 1 (standing technical committees);
- three years for level 2 (working parties or equivalent);
- two years for level 3 (expert groups)

They have been set in line with the anticipated length of the tasks that they are expected to deliver although adjustments can be made according to the programme of work.

3. CSNI, Joint Safety Research Projects and the Relationship with the NEA Databank

3.1 CSNI

The CSNI mandate is wide ranging in that it is responsible for the activities of the Agency that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations and within that responsibility is included the task of keeping all member countries involved in, and abreast of, developments in technical safety matters.

It has therefore, as shown in Figure 3, streamed its work into a structure of a bureau, five separate working groups, one of which has three subgroups (due to the diverse nature of its workstream) and three recent specific task groups that evolved from the NEA's work on Fukushima. Each group has its own mandate and activities that deliver when combined together the mandate of the CSNI.

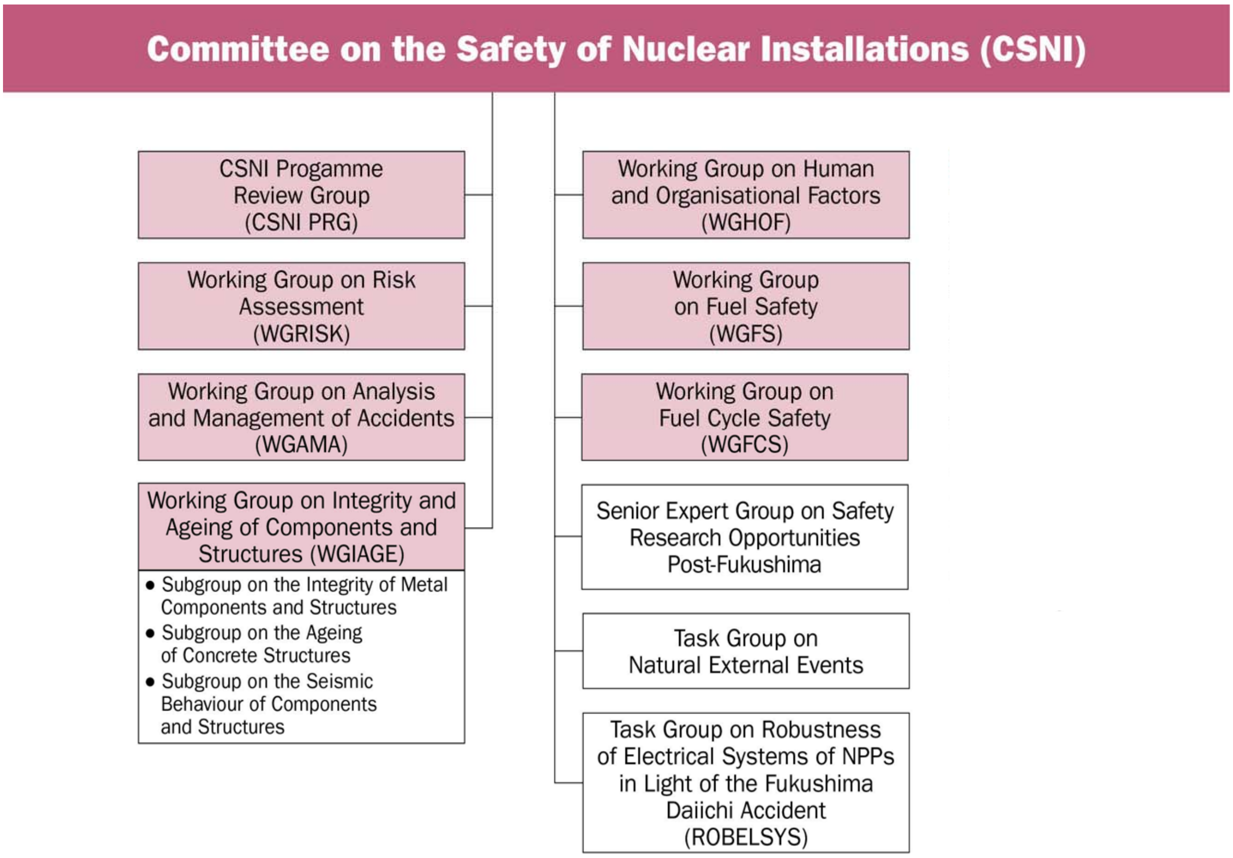


Figure 3. **Structure of CSNI**

3.2 Joint safety research projects

The Agency's Joint Safety Research Projects enable interested countries, on a cost-sharing basis, to pursue research or the sharing of data with respect to particular areas or problems. The projects are carried out under the auspices, and with the support, of the NEA.

The projects fall into two specific types, experimental and event database, currently there are running 13 safety research projects and 4 event database projects. Details can be found at this website (**http://www.oecd-nea.org/jointproj/**).

Experimental projects are initiated by a member country which funds typically 50% of the experimental work. The remainder is funded by other member countries and their organisations and interested non-member countries (within certain boundaries) provided there is a mutual benefit i.e. there must be contribution to the work as well as reception of the information.

The initiating member country shares its idea and seeks support from other member countries and interested parties. This support is traditionally gathered by sharing the idea at a number of the relevant working groups, seeking not only interest but expert advice to shape the experimental programme.

When an effective core of member countries has been gathered the CSNI gives permission for the setting up of an expert meeting to clarify the project and produce a more detailed experimental programme that is again circulated for interest and generates the likely contributors to the project.

Before commencing, the project must be approved by the full CSNI but once under way the project is governed solely by its signed agreement and its own appointed Management Board. The only requirements with regards to CSNI once it is approved are:

- When a project is finished it is asked to report the outcome to the working group which is traditionally by the issue of a final summary report.
- The data from the project is shared with the rest of the CSNI members once the initial non-disclosure period expires (set by the project members) in the interests of generally improving knowledge of nuclear safety.

One such experimental project of relevance to this this workshop is PRISME2. The second phase of the PRISME project (PRopagation d'un Incendie pour des Scénarios Multi-locaux Élémentaires or "Fire propagation in elementary multi-room scenarios") which provides valuable results to participants on room-to-room heat and smoke propagation, the effects of ventilation and the resulting thermal stresses to sensitive safety equipment in specific room configurations.

This project has been strongly supported by 12 member countries and has delivered significant data and knowledge to support the analysis of fire and smoke propagation in control rooms with open and confined situations.

Much more of interest to those involved in the Probabilistic Safety Analysis (PSA) field are the database projects.

These are:

- OECD/NEA Cable Ageing Data and Knowledge (CADAK) Project
- OECD/NEA Component Operational Experience, Degradation and Ageing Programme (CODAP)
- OECD/NEA Fire Incidents Records Exchange (FIRE) Project
- OECD/NEA International Common-cause Data Exchange (ICDE) Project

Unlike the experimental projects the database projects are groups of member countries consistently collecting and pooling data established within their own countries. The sharing, reviewing and processing of that data improves the overall knowledge base.

Data is often used to establish initiators for faults identify trends and anomalies and derive good practice that minimises failure and extends life of equipment and components.

These database projects are generally prolonged and of several phases with the data held by the NEA and accessed and updated on a project member country basis by national coordinators. As with the experimental projects this work can include non-NEA members but as with the experimental work this has to be only where there is mutual benefit.

The specifically relevant event database within this workshop is the FIRE project that has 12 participants from member countries and economies. The main purpose of the project is to encourage multilateral co-operation in the collection and analysis of data relating to fire events in nuclear reactor facilities and it has successfully become widely seen as the reference international database for fire events.

Two members of that Project will present a paper later in the workshop describing the use and applicability of the OECD NEA FIRE database project. This project has now entered its fourth phase during which it is intended that they will establish a mature long term development plan for maintaining and expanding the existing database and the direction of the analytical work.

3.3 Relationship between the research projects and NEA databank

The OECD NEA Joint Projects are managed using a common agreement between the participants. There is a commercial aspect as the participants all pay to fund the project. Thus, as part of the commercial process the data generated within the project is protected by an intellectual property clause and there is no general distribution of the data and knowledge to the rest of the CSNI.

However, as stated in section 3.2 above, for a project to be operated under the auspices of the OECD NEA it is necessary for it to be willing to share the data at some point with CSNI and indeed other NEA members in general.

This is achieved by the participants agreeing a non-disclosure date for the information to be generally released following the closure of the project. The period is set by the participants but the NEA Secretariat recommends to Project Management Boards that it is set as three years following the approval of the final report.

During the active period of a project and the non-disclosure period following the project closure it is possible to release and use some information by request to the Management Board. This is done via contact with the NEA Secretariat responsible for that project.

When a project closes all of the information generated during the active phase including papers and reports is gathered together and stored as a package within the NEA databank in the same way as the information from all of the other Joint Safety Research projects.

The goal of the NEA Databank is to be the international centre of reference for its member countries with respect to basic nuclear tools, such as computer codes and nuclear data, used for the analysis and prediction of phenomena in the nuclear field; and to provide a direct service to its users by developing, improving and validating these tools and making them available as requested.

The NEA Databank is therefore the most appropriate place to store the project data although the data itself does not form part of the contributions to the databank’s core work.

For individual data requests, during active project phases the NEA secretariat is contacted direct and liaises with project management board. Once closed the process is different. A requester should approach NEA via the NEA Databank website for Data Bank Computer Program Services (http://www.oecd-nea.org/dbprog/). There is then an internal NEA process for approval of release of data and other materials that recognises whether the request is for data within the non-disclosure period or for the data that is openly available to CSNI members etc.

The database projects are all in the active phase and thus requests for data are made through the NEA Secretariat and would not result in access to the raw data of the participants and although the decision is the Management Board's to take the NEA would recommend that release of any information should meet the mutual benefit test.

The processes described above relate to the requests and actions of individuals. The CSNI however encourages the projects to highlight the existence of new data and to consider sharing data with the working groups. There is a challenge therefore regarding extending the sharing of active project data with CSNI Groups

On the committee's behalf the NEA secretariat issued a letter in 2007 to all project Management Board chairs. The letter brought to their attention that the CSNI favoured a closer cooperation between CSNI Working Groups (WGs) and Joint Projects.

It explained that CSNI recognised that within projects there was a process that allowed the project Management Board to exceptionally approve the dissemination of their information outside the participating members within the period of three years after project completion.

CSNI therefore encouraged each project to:

- produce at least one international publication on the project so that others were aware of the work
- identify a point of contact if further information is desired
- further consider the transfer of general results from projects to working groups to support the development of working group programs (e.g., International Standard Problems: ISPs, State-of-the-Art Reports: SOARs).

The letter set down a process where if protected data was needed for performing an ISP or for producing a SOAR, a written request from the Working Group chair should be sent to the project Management Board chair. This written request should specify the motivation, scope and schedule for the WG activity. The Management Board of the project was then invited to consider that request and judge if release of information was acceptable.

4. Probabilistic Safety Assessment, WGRISK and Cross Cutting Issues

The NEA report "The Fukushima Daiichi Nuclear Power Plant Accident OECD/NEA Nuclear Safety Response and Lessons Learnt" (http://home.nea.fr/pub/2013/7161-fukushima2013.pdf) pointed out the increased degree to which PSA is being recognised by NEA member countries as a tool to assist in delivering the safety justifications for nuclear power plants. In addition the report recognised that there were ongoing discussions within member countries with the goal of identifying the best or optimum use of PSA in accident management.

CSNI as part of the NEA's post-Fukushima responses agreed to discuss the use of risk methodologies (PSA) for assessment of natural external hazards and a workshop was organised to initiate the task PSA of natural external hazards including earthquake.

The objectives of the workshop jointly held by WGRISK and the WGIAGE included to share methods and good practices and experiences in member countries on PSA analysis for natural external hazards and to identify potential new work in this area.

This workshop took place in October of 2013 and a report of the proceedings has now been produced for approval by CSNI. As with all of NEA work, the workshop coordinated experts in member countries to share information, discuss this information plus the methods and approaches being used and experience gained in PSA of natural external hazards.

The Workshop produce a number of conclusions including that while systematic approaches in PSA currently exist, additional work is needed. This work includes identifying more realistic evaluations to provide better view on the real problems. This type of work is an element of the work within the database projects as well as WGRISK one of the workshop's organising bodies.

These data projects can collect the types of information that can be useful in the development and review of PSA models. The WGRISK membership however, recognised there had been made little use of the data project products and thus initiated a task on "Use of OECD NEA Data Project Products in Probabilistic Safety Assessment" in NEA member countries in 2011. This task delivered a report NEA/CSNI/R(2014)2 that has been approved by the CSNI in December of 2013.

The objective of the task was stated as "strengthening of the relationships between the data project and PSA communities." and hopefully "increase industry support for the various OECD data projects by highlighting the potential benefits of these activities".

The final report highlighted how important the commitment and support of the member countries are to delivering a successful quality database for safety assessments. The report contains many recommendations but common themes include suggesting that CNRA and CSNI decision makers support further coordination, collaboration, and communication between the data projects and WGRISK as well as WGRISK with other working groups like the Working Group on operating Experience WGOE.

The main mission of the Working Group on Risk Assessment (WGRISK) is to advance the understanding and utilisation of probabilistic safety assessment (PSA) in ensuring the continued safety of nuclear installations in member countries. This can be considered part of a wider mission to provide risk-related support to the Committee on the Safety of Nuclear Installations (CSNI) as well as serving as an internationally recognized, authoritative source on risk-related matters and as an important resource for risk-related knowledge management activities.

Thus the mandate of WGRISK, along with the report on the use of data from the database, strongly supports the position that WGRISK should be considered a group that not only has its own inwardly facing work programme but also a portfolio of cross-cutting activities.

This is something the CSNI committee has indicated should be encouraged and the NEA secretariat intends to work with both the Chair of WGRISK and the chairs of the projects and other working groups to meet the desire to see WGRISK as a source of advice where risk-related support is required.

5. Relevant Structure of the NEA Secretariat, its Role and How to Access it

One of the main roles of the Nuclear Safety Division is to assist the two NEA safety committees, namely the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA).

The division co-ordinates the execution of their programmes as well as giving them and the working groups, advice, technical support and administrative assistance.

The staff of the division are highly experienced in one or more of the various areas of nuclear safety.

Their tasks are to:

- Assist and advise the chairpersons of the various committees and working groups,
- Help identify emerging issues,
- Assist the chairpersons in formulating proposals to address these issues,
- Bring together the representatives member countries representatives to discuss technical and procedural activities in meeting forums
- Co-ordinate the preparation and subsequent distribution of reports

Staff members are assigned responsibility for one or more working groups. The basic criterion used for making such assignments is the particular technical competence of the staff member. Short-term tasks or groups are assigned to staff members according to technical competence and workload.

The division is responsible for identifying the need for joint research projects, as well as to provide support to those projects agreed by member countries. Currently there are 13 safety research projects and 4 event database projects.

There are now 6 technical staff supporting the 12 CSNI activities and the 17 research projects. These can be identified and contacted through the webpages of the NEA safety division.

5. Conclusions

This paper has explained the structure of the NEA and the structure and work of the specific committees with safety mandates.

The paper has stressed the importance of member countries using the following to develop together best practice practical guidance in the field of nuclear safety and Fire PSA:

- Active involvement in working group activities
- Forums for the exchange of information and the transfer of experience
- Cooperation between member countries
- International co-operative activities of interest to the NEA member countries
- Mutually cooperation with the other standing committees on matters of common interest

The interrelationship between the CSNI, the joint nuclear safety research projects and the NEA databank has been explained as well as the methods by which experimental data can be acquired and shared.

The desire of the CSNI for each joint project to make others were aware of their work and to consider sharing that work has been reiterated and encouraged.

The efforts and progress that WGRISK has made in providing risk-related support CSNI as well as serving as an internationally recognized, authoritative source on risk-related matters and as an important resource for risk-related knowledge management activities have been highlighted.

The value in WGRISK offering a cross-cutting resource for advising CSNI activities has been expressed.

The methods by which NEA secretariat staff support the work of the members countries in the areas of safety research and delivering best practice guidance have been drawn out. Particularly, how they provide advice and guidance to the member countries and the representatives as they deliver their work programmes.

Finally it is important to note that it is always the intention of the NEA secretariat to ensure that the members are brought together in mutual cooperative endeavours to deliver best practice practical guidance in many areas including the field of nuclear safety and Fire PSA.

Fire PRA Maturity and Realism: A Technical Evaluation and Questions

N. Siu and S. Sancaktar
U.S. Nuclear Regulatory Commission, USA

Abstract

Fire PRA has often been characterized as being less mature and less realistic than internal events PRA. Perceptions of immaturity can affect stakeholders' use of fire PRA information. Unrealistic fire PRA results could affect fire-safety related decisions and improperly skew comparisons of risk contributions from different hazards. This paper addresses the issue of technical maturity through the identification of a number of key indicators. It addresses the issue of realism primarily through a number of quantitative and qualitative comparisons of fire PRA results with operational event data. Rather than attempting to resolve the ongoing debate on the maturity and realism of fire PRA, this paper offers a few current thoughts from a work in progress to clarify some of the terms of the debate, brings some additional evidence to the discussion, and raises pertinent questions.

1. Background

1.1 Estimated risk importance of fire

Since the earliest industry-sponsored full-scope probabilistic risk assessments (PRAs) (e.g., the Indian Point Probabilistic Safety Study, reviewed for the U.S. Nuclear Regulatory Commission – NRC – in Ref. 1), and continuing through the NRC's NUREG-1150 [2] and Risk Methods Integration and Evaluation Program (RMIEP) studies [3] and the industry's Individual Plant Examinations of External Events (IPEEEs) [4], fire has been shown to be a significant risk contributor for U.S. plants.[1] This finding, which is illustrated by the core damage frequency – CDF – metrics provided in Table 1, is not unique to the U.S.; international studies also recognize the risk importance of fire [5, 6].

In 2004, the NRC modified its fire protection rule (10 CFR 50.48) to provide licensees with a voluntary, risk-informed option for meeting the NRC's fire protection requirements. This rule change endorsed the National Fire Protection Association (NFPA) Standard 805 "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (commonly referred to as "NFPA 805") [7]. Several licensees have submitted Licensing Amendment Requests (LARs) to take advantage of this change. The LAR submittals largely employ the fire PRA guidance documented in the joint report EPRI 1011989/NUREG/CR-6850 (henceforth referred to as "NUREG/CR-6850") [8], and a supplement to that report capturing lessons learned from pilot submittals [9]. The LAR submittals indicate that the estimated fire CDFs remain significant (see Table 2). The last line in Table 2 is noteworthy, particularly in comparison with the last line in Table 1. We will return to this point later.

[1] *This finding applies to the general population of plants. Whether fire is an important contributor for a particular plant (and what are the dominant risk scenarios) is very much a function of plant-specific details.*

Table 1. **CDF estimates from early U.S. fire PRAs**

Study	Plant(s)	Mean Fire CDF (/ry)	% Fire Contribution to Total CDF[a]
Indian Point (1982)	Indian Point 2	2.0E-4	38
	Indian Point 3	9.9E-5	50
NUREG-1150 (1990)	Surry 1	1.1E-5	14[b]
	Peach Bottom 2	2.0E-5	72[b]
RMIEP (1992)	LaSalle 2	3.2E-5	32
IPEEE (mid-late 1990s)	99 units	3.7E-5[d]	26[c,d]

[a]Computed as the ratio mean fire CDF/mean total (all hazard) CDF; both CDFs are for at-power conditions
[b]Total CDF computed using seismic CDF based on Electric Power Research Institute (EPRI) seismic hazard curves
[c]Population average
[d]Computed for plants performing seismic PRAs (not seismic margins assessments)

Table 2. **Summary statistics from a representative sample of NFPA 805 LARs**

Sample size	15 units
Submittal dates	2011-2013
Average reported fire CDF	3.9E-5/ry
Minimum reported fire CDF	6.5E-6/ry
Maximum reported fire CDF	6.5E-5/ry
% fire contribution to total CDF	68

1.2 The problem

Despite a history nearly as long as that for internal events analysis, fire PRA has often been characterized as being less mature and less realistic than internal events PRA [10-13]. Gallucci [14] provides in-depth documentation and analysis of on-the-record statements (many from public meeting transcripts) regarding the issue of fire PRA maturity and conservatism. Some of these statements assert (as above) that fire PRA methods are relatively immature and produce conservative results, while others provide a moderating or even contrary point of view.

The two related (but non-identical, as discussed in the following section) issues of fire PRA maturity and realism are important practical matters. Following the NRC's 1995 PRA Policy Statement [15], PRA results and insights are being increasingly used in regulatory applications to the extent supported by the PRA state of the art. These applications range from plant-specific (e.g., the management of plant maintenance activities, the approval of changes to a plant's licensing basis, the assessment of the significance of inspection findings) to industry-generic (e.g., the assessment of potential safety issues affecting more than one plant, the determination as to whether new regulatory requirements should be imposed on the industry). Depending on the particular application, a variety of PRA outputs, including importance measures, accident frequencies (both CDF and large early release frequency – LERF), changes in accident frequencies, and relative contributions to risk may be called for. Clearly, if the analysis of a risk-significant hazard (or hazard group) is unrealistic, the PRA could be providing faulty information to the decision making process. Moreover, an unrealistic analysis could skew comparisons of risk contributions from different hazards, thereby distorting our understanding of risk and degrading one of the major benefits of PRA, which is to help focus attention on areas of "true safety significance" [15]. Even further, if the PRA analysis of a hazard is viewed as immature (or less mature than analyses of other important hazards), stakeholders might be tempted to overly discount even useful information from the PRA in lieu of evidence from other sources (e.g., global statistical estimates, worst case analyses) that may have their own, if less thoroughly examined weaknesses.

In this paper, we will not attempt to resolve the ongoing debate on the maturity and realism of fire PRA. Rather, we offer a few current thoughts from a work in progress to clarify some of the terms of the debate, bring some additional evidence to the discussion, and raise pertinent questions.

2. Maturity and Realism in a PRA Context

As seen from the previous section, the issues of fire PRA maturity and realism are often raised in concert. We believe that although related, they are actually separate. The concept of maturity addresses the relative state of development of a technical discipline. On the other hand, in a PRA context, the concept of realism addresses the degree to which an analysis represents the current state of knowledge relevant to the decision problem.[2] The analytical technology (i.e., methods, models, tools, and data) of a less mature discipline could, but need not, produce unrealistic analysis results. Conversely, a more mature discipline could, for practical reasons, employ technology with known weaknesses, only requiring that the weaknesses be understood and appropriately addressed in the decision making process.[3] Of course, the practitioners of a less mature discipline might consciously use conservative (and unrealistic) assumptions in an attempt to compensate for weaknesses in the current state of knowledge – the extent and appropriateness of this practice is the key controversy in ongoing U.S. fire PRA applications[4] – but this observation only shows that the issues are coupled, not identical.

3. On the Maturity of Fire PRA

Judging the maturity of a technical field is a subjective matter, being dependent on the judgment of the assessor. To add some structure to the debate, we note that Stetkar and his co-authors are careful to distinguish between the maturity of the fire PRA technology (which dictates what level of analysis is possible) from the maturity of the application of that technology (which indicates what is happening in the field) [12]. They also tie the notion of maturity to the number of experienced analysts performing fire PRAs. Budnitz provides similar indicators in a discussion of the state of seismic PRA [19], referring to the number of practitioners (or groups of practitioners), the degree of practice, and the state of technical development of the field (including the availability of detailed guidance for new practitioners). Budnitz emphasizes the use of the technology in support of practical decision making as an important indicator of maturity. This

[2] *In his 2003 speech "Realism and Conservatism," then-NRC Chairman Diaz defines the term "realistic" as "being anchored in the real world of physics, technology and experience" [16]. In a PRA context, because a) the PRA needs to deal with rare (and hopefully unobserved) events, and b) the purpose of the PRA is to support decision making, we think it appropriate to tie the notion of "realism" to the needs of decision making.*

[3] *For example, NUREG-1855 [17] advocates the use of consensus models coupled with sensitivity analyses to address key model uncertainties.*

[4] *Interestingly, the Lewis Commission's 1978 review of the seminal 1975 Reactor Safety Study (WASH-1400) also raised a concern with "a pervasive regulatory influence in the choice of uncertain parameters" [18].*

emphasis is echoed in a Technical Opinion Paper issued by the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations (NEA/CSNI) [20]. Finally, in a thoughtful exposition on the state of structural safety engineering, Cornell describes characteristic situations associated with the different stages of development of a technical field based on his observations from a number of fields (including geotechnical engineering, structural dynamics, and finite element analysis) [21].

Table 3 provides a summary of Cornell's discussion, grouping his situations into one of three categories of indicators involving the field's practitioners, research agenda, and applications.

Table 3. **Indicators of Stages of Technical Maturity (adapted from Cornell [21])**

	Developmental Stage		
	Early (Infancy, Emerging)	Intermediate (Adolescent, Developing)	Late (Mature, Stable)

Practitioners	• Small research community
	• Small number of practitioners
	• Strong personality influences, competing schools of thought
		• Larger number of practitioners
		• Larger number of experienced researchers
			• Many well-trained and experienced practitioners
			  • Recognize limits of applicability of methods
			  • Can adapt methods to new situations
			  • Can work with researchers to identify important issues

Research Agenda	• Driven by perceived needs • Problem selection affected by personal choice (e.g., due to ease of formulation or solution)	• New practice-driven research problems • Some consensus positions for some broadly defined problem areas • Some unproductive research lines abandoned • Incomplete coverage of topics	• Most research driven by needs of practice • More abstract research addresses needs clearly identifiable by all concerned
Applications	• Local applications (addressing small parts of larger problems) • No broader framework	• Fast growth • Developing vocabulary • Optimistic views on new methods; limitations not well understood	• Vocabulary has evolved • General framework exists • Little "selling" of area

Applying the preceding ideas to our experience in developing and applying fire PRA methods, models, tools, and guidance, it appears to us that nuclear power plant fire PRA is: a) in an intermediate stage of development (but well past the early stage), and b) less developed than internal events PRA. A key factor in the first part of our assessment is the acceptance of fire PRA results in supporting major decisions, starting with the Commission's 1985 decision to allow continued operation of the Indian Point Plants [22] as informed by findings and recommendations of the NRC's Atomic Safety and Licensing Board (ASLB) [23],[5] continuing with plant changes identified in the IPEEE program[6] [4] and more recently with staff approvals of licensee-requested fire protection program transitions as per NFPA 805.[7] Key factors in the second part of our assessment are the relatively small number of fire PRA practitioners (as compared with internal events) and the lack of consensus models and data for a number of important issues. We recognize that, as pointed out by Stetkar et al. [12], the ongoing licensee and staff activities related to NFPA 805 will increase the fire PRA experience base, and will likely, over time, reduce the maturity gap with internal events.

Of course our assessment is subjective; others can review the available information and reach a different conclusion. Given that the issue of maturity tends to be self-resolving as long as there are practical application needs, perhaps such differences of opinion shouldn't matter very much. However, should discussion be needed, or, more practically, should we wish to accelerate the maturation process, we suggest that a structured consideration of indicators such as those we've identified above is useful. We note that these indicators suggest several possible actions one could take to increase the maturity of a field – research and development aimed at improving the analytical technology is only one such action. The indicators also support the point made by Stetkar et al. [12] and others (see, for example, the quotes provided by Gallucci [14]), that substantial changes in fire PRA maturity are likely to take many years; there is no "quick fix."

4. On the Realism of Fire PRA

Fire PRA, as with PRA in general, is aimed at identifying risk-significant scenarios and quantifying their likelihoods and consequences. In principle, it can address scenarios with a wide range of consequences (e.g., various states of plant damage). In practice, the analytical resources of U.S. fire PRAs are typically focused on scenarios leading to core damage and (in recent times) large, early release. To accomplish this, the analysis, as originally formulated and currently practiced, is iterative [25-27]. Potentially important scenarios are identified, conservatively assessed, and passed on to more detailed analysis stages if they meet certain screening criteria. The intent is that the overall results of the analysis be sufficiently realistic for the purposes of the study; there is no guarantee that the analyses of non-contributing scenarios, some of which may be important contributors to intermediate end states (e.g., loss of specified safety functions but not core damage), are realistic.

The strong tie of the analysis results to the specific purpose of the analysis complicates our assessment of realism. In this section, we look at this topic from a number of angles: the summary and detailed outputs of past and recent fire PRAs, and the technology (i.e., the methods, models, tools, and data) of fire PRA.

4.1 Fire CDF estimates

One natural approach to assess the realism of fire PRA is to compare its summary output measures (notably, fire CDF) against appropriate empirical benchmarks, e.g., statistical estimates derived from operational experience. However, such a comparison is not straightforward.

[5] *In the Indian Point PRA, which played a major role in the ASLB hearing, fire was shown to be a major contributor to CDF (see Table 1). The ASLB, in its remarks on PRA areas needing modelling improvement, mentioned fire as one of a number of areas needing improvement (the others were the treatment of operator diagnosis of accidents in progress, DC power supply failures, common mode failures due to plant maintenance, seismic hazard, and hurricane and tornado hazard), but did not place special emphasis on this point [23].*

[6] *Although the NRC has not tracked changes identified in the IPEEE studies, a number of IPEEE submittals indicated that changes had actually been made [4].*

[7] *As of this writing, the staff has approved five NFPA LARs and an additional 22 LARs are under review [24].*

4.1.1 Challenges in comparing Fire PRA CDF estimates with statistical estimates

From a practical standpoint, data for severe fire events are sparse. A review of international operating experience documented in NUREG/CR-6738 [28] shows that although there have been some “close calls” to core damage,[8] there have been no actual core damage events caused by fires of interest to U.S. fire PRAs.[9] A review of operational events reported to the NRC via Licensee Event Reports (LERs) and analysed under the NRC’s Accident Sequence Precursor (ASP) program [29] indicates that in the period 1980-2012, only around 80 have been initiated by (or later involved) fires.[10] The vast majority of these did not represent major challenges to nuclear safety: none were classified as “significant” (with Conditional Core Damage Probabilities – CCDPs – greater than 1E-3) and only two had CCDPs between 1E-4 and 1E-3.[11] Out of the 1695 reviewed fire events for the period 1990-2009 included in the EPRI Fire Events Data Base, only 28 were classified as “challenging,” and this designation is based on the severity of the fire severity, i.e., its ability to damage components, but not the nature or significance of the components actually affected [30].

From a theoretical standpoint, simple statistical analyses for CDF of the sort seen in the literature following the March 11, 2011 Fukushima Dai-ichi reactor accidents (e.g., [31-33]), which involve dividing the number of observed core damage events by the number of reactor years, are based on two underlying assumptions: a) the plants in the analysis group are nominally identical, and b) the plants do not change over time. Apostolakis [34], who refers to these assumptions under the unifying title of “exchangeability,” argues that from a regulatory decision maker’s perspective, both assumptions are questionable. Regarding the first, one of the early, fundamental lessons from PRAs over the years is that risk is plant specific [35]. Regarding the second, U.S. plants have made numerous fire-safety related improvements in response to events and associated regulatory actions (e.g., the promulgation of Appendix R to 10 CFR 50.48 following the Browns Ferry fire) and analyses (e.g., the IPEEEs). More sophisticated statistical analysis techniques are available to address heterogeneity within a group and time dependence. However, such methods require even more data than the simple approach described above.

To provide a general indication of the potential magnitude of variability across plants and over time, we plot recent point estimates of total CDF (i.e., the CDF from all contributors) obtained from 41 risk-informed LARs (covering 61 units) submitted over the period 2002 through 2013 (over 75% were submitted after 2007) in Figure 2. Figure 3 shows how recent estimates for total CDF compare against estimates derived from the IPE and IPEEE studies.[12]

---

[8] *In addition to the well-known 1975 Browns Ferry cable fire, NUREG/CR-6738 identifies five fire events that seriously challenged nuclear safety: Greifswald 1 (1975), Beloyarsk 2 (1978), Armenia 1&2 (1982), Chernobyl 2 (1991), and Narora 1 (1993). As discussed later in this paper, all of these events involved fire-induced loss of multiple safety systems; three (Greifswald, Armenia, and Narora) involved station blackout conditions either caused by the fire or by fire-fighting actions.*

[9] *The 1957 Windscale accident involved a fire in the reactor’s graphite-moderated core. The 1979 Chernobyl 4 accident also involved a graphite fire – this fire was the result rather than the cause of the reactor power excursion which damaged the core.*

[10] *The NRC’s LER database can be accessed via https://nrcoe.inel.gov/secure/lersearch/index.cfm.*

[11] *The statuses of the ASP program and the Standardized Plant Analysis Risk – SPAR – models used to estimate event CCDPs are described in annual NRC staff papers, e.g., SECY-13-0107 [33]. The ASP program considers all events and degraded plant conditions reported in LERs per the requirements of 10 CFR 50.73. Currently an event is classified as a precursor if the CCDP for that event is greater than a set criterion (the greater of 1E-6 and the plant-specific CCDP for a non-recoverable loss of balance-of-plant systems).*

[12] *Figure 3 is based on the results from plants that: a) recently submitted a risk-informed LAR that addresses CDF contributions from all initiators, and b) performed a seismic PRA as part of their IPEEE analysis.*

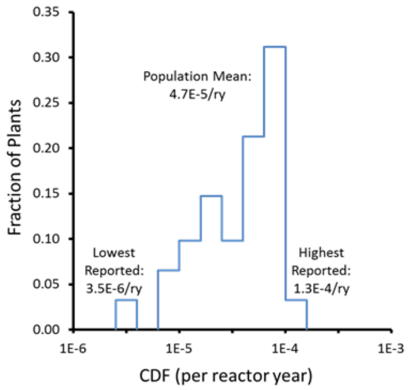


Figure 2. **Recent total CDF estimates**

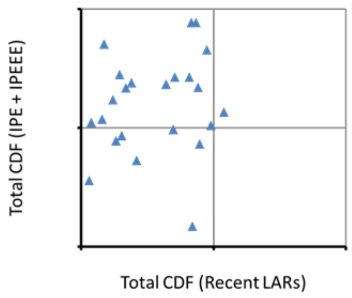


Figure 3. **Comparison of recent and past CDFs**

Figure 2 shows that most of the total CDF estimates fall between 1E-5/ry and 1E-4/ry. Figure 3 shows that most (but not all) of the total CDFs have decreased, some by a substantial amount. Based on a review of some of the LAR submittals, it appears that these changes can be attributed to both modelling changes and the incorporation of actual improvements in plant design and operation.

We caution both figures and our following analysis are provided only for the purpose of a rough comparison. The LAR estimates were developed for varying purposes and are of varying vintage, and may therefore embed different differences in modelling assumptions, level of detail, and perhaps even scope.

4.1.2 Empirical and Fire PRA CDF estimates

Without challenging decision maker concerns regarding the exchangeability of events across the U.S. fleet and over time, we think that, for the purpose of discussing the realism of fire PRA, it's useful to explore how the predictions of current fire PRAs compare against available, plant-level statistical evidence. In particular, does the statistical evidence support or deny assertions of fire PRA conservatism?

To perform this comparison in the presence of sparse data, we follow the approach of Gallucci, who uses event precursor CCDPs developed by the ASP program as data points [36]. Based on precursor events covering the period 1969-2004 (see Table 4), Gallucci estimates that the average fire CDF for a U.S. plant is 7.1E-5/ry.

Table 4. **Precursor events included in Gallucci's analysis [37]**[13]

Plant	**Date**	**CCDP**	**Event Notes***
Browns Ferry 1 & 2	3/22/75	0.20**	Multi-unit cable fire; multiple systems lost, spurious component and system operations; makeup from control rod drive pump (non-proceduralized action)

[13] *A number of the listed CCDP estimates differ from those provided in Refs. 39 and 40. Also, the Watts Bar Hydroelectric Station fire is not relevant for our discussion. However, as pointed out by Gallucci, the analysis results are dominated by the CCDP value assigned to the Browns Ferry fire; the values of the other CCDPS have negligible effect.*

Plant	Date	CCDP	Event Notes*
Rancho Seco	3/19/84	2.2E-6	Main generator explosion and fire; damage to non-nuclear instrumentation power supply complicated shutdown
Oconee 1	1/3/89	3.3E-6	Reactor coolant pump (RCP) switchgear fire during power escalation; operators exceeded allowed cooldown rate
Waterford 3	6/10/95	9.1E-5	Non-safety 4 kV switchgear fire; partial loss of offsite power (LOOP)
Surry 1 & 2	10/9/99	1.2E-6 (each)	4 kV bus bar connection fire (small); loss of two emergency buses due to electrical fault
Diablo Canyon	5/15/00	9.6E-5	12 kV bus fire damaged nearby 4 kV bus; loss of offsite power to all 4 kV loads
San Onofre 3	2/3/01	1.4E-4	Switchgear fire following outage; loss of non-safety power
Quad Cities 2	8/2/01	6.6E-5	Main transformer fire following lightning strike; loss of normal offsite power
Watts Bar 1	9/27/02	3.3E-4	Offsite (hydroelectric station) fire; LOOP, fire brigade dispatched offsite, reduced onsite staffing

*See NUREG/KM-0002 [37] for a recent compilation of information on the Browns Ferry fire and NUREG/CR-6738 for a fire-PRA oriented discussion of that event. The notes on the remaining events are based on their LER Summaries.
**Consistent with SECY-10-0125 [38]. Gallucci notes other estimates range from 0.03 to 0.40.

Table 5. **Post-2004 fire precursors with CCDP > 1E-4**

Plant	Date	CCDP [29]	Event Notes
H.B. Robinson 2	3/28/10	4E-4	4 kV cable fire, loss of RCP seal cooling and additional equipment failures; operators fail to diagnose plant conditions and control plant; operator actions cause a second fire
Fort Calhoun	6/7/11	4E-4	480V switchgear fire during cold shutdown; loss of multiple safety buses (combustion products migrated to non-segregated bus duct)

In the years following Gallucci's analysis, two important fire-related precursor events (but no significant fire-related precursor events) have occurred in the U.S. These involved a March 28, 2010 fire at the H.B. Robinson 2 plant, and a June 7, 2011 fire at the Fort Calhoun plant (see Table 5).

To update and extend Gallucci's analysis, we:

(1) incorporate the post-2004 operating experience, including the Robinson fire[14] (but not the Fort Calhoun fire since our analysis addresses at-power conditions);

(2) perform a Bayesian analysis to develop a distribution for the average plant fire CDF to quantify the uncertainty in the estimate;

(3) use the result of the Bayesian analysis to estimate the distribution of F-CDF_{US}, the total U.S. fire CDF (i.e., the sum of all of the individual plant fire CDFs)[15]; and

[14] *We do not include the Fort Calhoun event since the expressed concerns with fire PRA (and hence our analysis) are centered with PRA for at-power conditions. From a strictly numerical perspective, the inclusion of the Fort Calhoun event would not significantly affect our results.*

(4) compare this precursor-based distribution for F-CDF_{US} against a distribution for F-CDF_{US} derived from recent NFPA 805 LAR submittals.

Our Bayesian assessment in Step (2) uses Gallucci's point value estimate as the mean value of a constrained non-informative prior distribution [41]. We compare precursor- and PRA-based sums rather than averages in Step (4) to ensure an "apples to apples" comparison (the precursor-based estimate addresses an "average plant," whereas the PRA estimates are plant-specific) and to facilitate comparisons with total U.S. operating experience.

Based on our analysis, the mean value of F-CDF_{US} is about 6E-3/yr. (This is slightly lower than Gallucci's result because no significant fire precursors have occurred since his analysis.) Comparing this result with the mean value of F-CDF_{US} derived from the NFPA 805 LAR submittals (about 4E-3/yr), it can be seen that the fire PRAs would appear to provide a smaller estimate than the precursor-based analysis. However, when we plot the distributions of F-CDF_{US} (see Figure 4), we see this comparison of the mean values can be misleading. The uncertainties in the precursor-based estimate are extremely large (due to the weakness of the operational evidence), and so the comparison of F-CDF_{US} estimates actually does not provide a definitive statement regarding the conservatism (or non-conservatism) of current fire PRA results.[16]

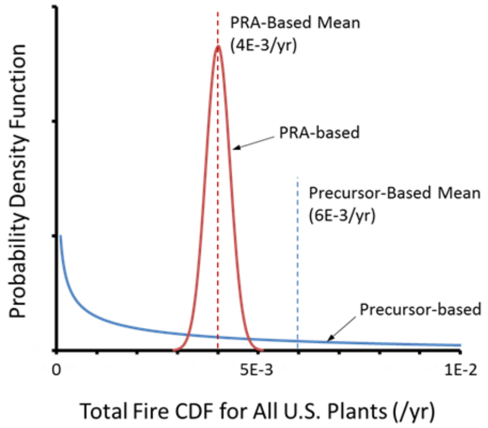


Figure 4. **Comparison of precursor- and PRA-based distributions for F-CDF_{US}**

Figure 4 shows quite different states of knowledge resulting from the two different sources of evidence. To explore the significance of this difference, consider the probability of observing N fire-induced core damage accidents (anywhere in the U.S.) over a time period T (where N = 0, 1, 2, etc.). This probability is the Poisson distribution averaged over all possible values of F-CDF_{US}:

$$P(N|T)=\int_0^\infty \frac{(f\cdot T)^N}{N!}e^{-f\cdot T}\pi(f)df \qquad (1)$$

[15] *Conceptually, this step involves the multiplication of the Bayesian result for a single plant by the number of operating U.S. plants (roughly 100).*

[16] *The PRA-based distribution treats the uncertainties in the plant PRA estimates. However, the distribution is quite narrow because it represents the sum of these estimates.*

In this equation, which substitutes the symbol "f" for F-CDF_{US} for simplicity, $\pi(f)$ is the probability density function for F-CDF_{US}.

Figure 5 shows the results of Eq. (1) for both the precursor- and PRA-based cases when T is set to 10 years. It can be seen that the differences between the precursor- and PRA-based estimates are negligible. (The difference is most noticeable for N = 2, an unrealistic situation since should a core damage event actually occur, major changes to plants, the regulatory system, etc., and thereby F-CDF_{US}, will almost certainly result.) Similar conclusions result even when T is set to 50 years.

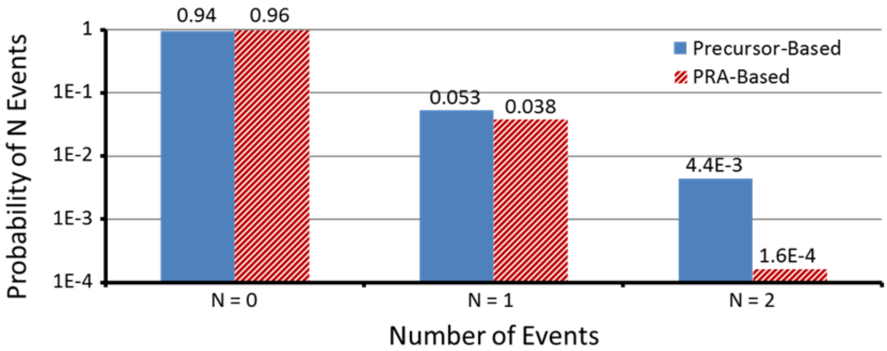


Figure 5. **Comparison of precursor- and PRA-based estimates of U.S. fire-induced core damage event probabilities**

We caution that our analysis:

- relies on the questionable assumption of exchangeability;
- is limited to precursors that involved initiating events (e.g., a plant trip or LOOP) – it does not address the CDF implications of precursors involving degraded conditions;[17]
- uses event CCDPs that quantify the possibility of "what-ifs" associated with plant response to an initiating event (e.g., additional independent hardware failures) but do not address different possibilities associated with the triggering fire (e.g., different locations or severities) – the observed fire-induced damage is a "given;" and
- produces results that are somewhat sensitive to the assessed CCDP for the Browns Ferry fire. (For example, the precursor-based probability of N = 1 changes to 0.011 if the CCDP is 0.03, and to 0.083 if the CCDP is 0.40.)

With these caveats, our results do not provide a "smoking gun" supporting assertions of fire PRA conservatism.

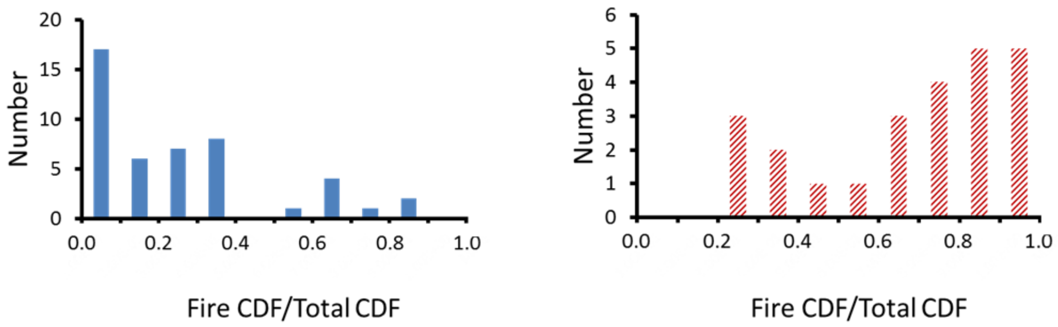
4.1.3 Relative contributions to total plant CDF

The preceding analysis uses available operational experience but requires a number of assumptions, the most important one being that of event exchangeability. To provide a second, but still CDF-

[17] *SECY-13-0107 uses an "integrated ASP index" to address both initiating events and plant conditions but notes the difficulty in estimating CDF from this information.*

based perspective on the realism of current fire PRAs, we look at past and current estimates for the relative contribution of fires to the overall CDF.

Figure 6 compares the relative contribution of fire to total CDF from the IPE/IPEEE studies (mainly performed in the mid-late 1990's) and from recent (post-2007) risk-informed LAR submittals. The IPE/IPEEE results come from the 46 plants which either completely screened seismic events or developed seismic CDF estimates. The 24 LAR estimates primarily involve NFPA 805 plants, but a few involve other risk-informed applications (e.g., plant Technical Specification modifications). Figure 7 compares the ratio of fire CDF to internal events CDF for the IPE/IPEEE studies (98 plants) and for the same set of LAR submittals addressed in Figure 6b.



(a) IPE/IPEEE **(b) Recent LAR**

Figure 6. **Fire contribution to CDF: comparison of IPE/IPEEE and recent LAR results**

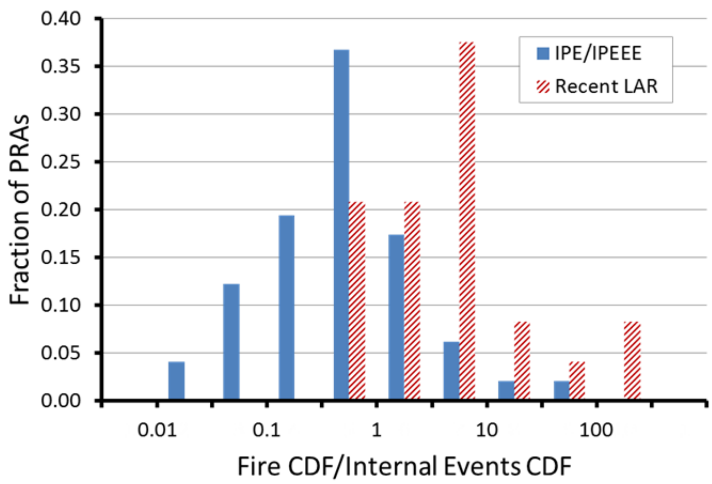


Figure 7. **Ratio of fire CDF to internal events CDF: comparison of IPE/IPEEE and recent LAR results**

Recognizing that the recent LAR submittals represent a smaller sample, nevertheless the difference between the two sets of results is striking. In the IPE/IPEEE studies, fire is an important contributor for many plants. In the recent LAR submittals, fire is a major or even dominant contributor for most plants. Possible explanations for this change include: a) the numerous plant changes made since the IPE/IPEEE studies were preferentially effective for non-fire related initiators (a difficult proposition, given the importance of plant response to fire risk), b) the IPEEE studies underestimated the importance of key issues addressed in the recent studies (we discuss changes in fire PRA technology later in this paper), or c) the recent fire PRA results are indeed conservative.

4.2 Important scenarios

Similar to our analysis of fire CDF, it's interesting to compare fire PRA scenarios with scenarios from actual operational experience. Such a comparison cannot provide definitive conclusions because: the empirical data are sparse (and many of the potentially relevant events are quite old, pre-dating many important plant improvements), the fire PRA identifies a myriad of possibilities, and even low-likelihood events can occur. Nevertheless, we qualitatively explore whether:

1) important fire PRA scenarios been observed in major fire events, and,
2) major fire events have involved scenarios not typically addressed by fire PRAs.

4.2.1 Fire PRA scenarios

Past studies (including the IPEEEs), taken as a whole, have consistently found that fires involving electrical cables and/or cabinets in key plant areas (e.g., main control rooms, emergency switchgear rooms, cable spreading rooms, cable vaults and tunnels) are the dominant contributors to fire risk [1-4, 42, 43]. In a number of these areas, the risk-significant scenarios can involve fires that start in electrical cabinets but propagate to cables outside. Typically, the fire effects are relatively localized (i.e., not room-encompassing) – the fire is important because it affects a local concentration of important cables. However, the IPEEEs have shown that for some plants, large turbine building fires and fires inducing main control room abandonment could be important. The risk-significant accident sequences triggered by fires are generally dominated by some form of transient (e.g., loss of feedwater, LOOP, loss of various support systems) but loss of coolant accidents (LOCAs), including reactor coolant pump (RCP) seal LOCAs and transient-induced LOCAs involving stuck open relief valves are important for some plants. Scenarios involving non-fire related failures can be visible contributors to risk, but the risk tends to be dominated by scenarios in which the initiating fire causes enough damage to cause core damage directly (if such scenarios exist for the plant being analysed) [42, 44].

The technical lessons stemming from recent fire PRA studies have not yet been synthesized as most of the NFPA 805 submittals are undergoing NRC review. To shed some light on important scenarios, we consider the results of NRC's SPAR-AHZ models[18] [29, 45]. The three most recent models (all for PWRs) address fire scenarios using information from NFPA 805 submittals. These models are benchmarked against the licensee models; the differences are not important for the purposes of this paper.

The important scenarios identified by the three SPAR-AHZ fire models are, for the most part, consistent with those identified in past studies. Electrical fires in the usual important areas (e.g., main control rooms, cable rooms, switchgear rooms) and turbine building fires are important contributors at some or all of the plants. The plant response scenarios triggered by these fires typically involve some form of transient (including LOOP scenarios), sometimes involving the

[18] *The NRC's SPAR models are used to support a number of staff activities, notably the assessment of the significance of operational events and of inspection findings. The SPAR-AHZ – "All Hazard" – models, similar to the older SPAR-EE ("External Event") models developed over the period 2005-2010, are intended to enable integrated, plant-specific analyses of internal hazards (including equipment failures, human errors, and internal floods) and external hazards using a single SPAR model.Currently, 20 SPAR-AHZ models have been developed.*

spurious opening of a power-operated relief valve (PORV). Fire-induced RCP seal LOCAs are not important contributors at these plants.

The greatest difference between the SPAR-AHZ models and older studies concerns yard fires. In the SPAR-AHZ models, these fires (which include fires involving large station transformers), are either the top or the number two contributor for the three plants.

Some additional observations concerning the three SPAR-AHZ fire model scenarios are as follows.

- The total frequency of scenarios involving reactor trip (automatic or manual) ranges from 0.06/ry to 0.30/ry. (As with past fire PRAs, it is assumed that every unscreened fire scenario results in a reactor trip.)
- The total frequency of scenarios involving fire-induced LOOP ranges from 8E-3/ry to 1.5E-2/ry. These LOOPs are modelled as being unrecoverable.
- Scenarios involving main control room abandonment are not major contributors, ranging from 0.1% to 2% of total fire CDF. The CCDPs for these scenarios span a large range, going from 0.06 up to 1.0. For plants with higher CCDPs, it can be seen that the low CDF contribution is due to the estimated low frequency of fires spurring evacuation, not the modeled robustness of the plant response.
- The fire PRA models generate thousands of detailed event sequences that need to be quantified. (By comparison, the older SPAR-EE models generate on the order of 50 sequences to be quantified, more for models if MCR scenarios are divided into cabinet-level sub-scenarios.) This creates challenges not only for the software quantification tools, but also more subtle challenges for model checking during model development and after quantification.

4.2.2 Observed fire scenarios

Tables 4 and 5 list notable U.S. fire precursor events occurring in the period 1969-2012. Other than the 1975 Browns Ferry fire, none of these involved multiple safety system losses and serious challenges to core cooling.

Table 6 provides summaries of the five non-U.S. fire events involving multiple safety system losses and serious challenges to core cooling identified and analysed in NUREG/CR-6738. It is important to recognize that none of these events involved plants of U.S. design, and that the latest event occurred in 1993; we are unaware of any severely challenging fires since the Narora fire.

Table 6. **International Fires Involving Severe Challenges to Core Cooling**

Plant	Type	Date	Summary Description (based on narratives provided in Ref. 28)
Greifswald 1	VVER-440	12/7/75	92 min cable fire in or near 6 kV switchgear started by electrical fault; caused station blackout (SBO), loss of all normal core cooling for 5 hours, loss of coolant through pressurizer safety (failed to reclose); recovered through low pressure pumps and cross-tie with Unit 2 to power one AFW pump (non-proceduralized actions).
Beloyarsk 2	LWGR -1000	12/31/78	Lube oil fire in turbine building collapsed turbine building roof, propagated into several elevations of control building (open penetrations, cable shafts); damaged main control room (MCR) panels; secondary fire from oil-filled transformer; extreme cold weather, fire under control in 17 hours, extinguished in 22 hours; damage to multiple safety systems and instrumentation, reactor control was “extremely difficult.”

Plant	Type	Date	Summary Description (based on narratives provided in Ref. 28)
Armenia 1 & 2	VVER-440	10/15/82	Short circuit in a 6 kV cable led to 7 fires (ignition points) in 2 different cable galleries; fire spread to other cables, smoke spread to several areas including Unit 1 MCR; Unit 2 had some lesser fire effects; automatic foam system in manual and not actuated, fire brigade did not start attack for 20 minutes because power was on; fire under control in 6 hours, extinguished 1 hour later; Unit 1 SBO caused by hose streams about 2 hours into fire, loss of instrumentation and reactor control about 1 hour later; event also involved secondary H_2 explosion, lube oil fire and transformer explosion; recovered via temporary cable from emergency diesel generator (EDG) to high pressure pump (non-proceduralized action), recovery of Unit 1 MCR power from Unit 2 sources.
Chernobyl 2	RBMK-1000	10/11/91	Turbine failure, H_2 and oil release, large turbine building fire, turbine building roof collapsed causing loss of generators, eventual loss of all feedwater (direct damage from falling debris or de-energization to aid local fire fighting; feedwater supply intermittent, operators use seal water supply system for makeup; reactor control regained in 3.5 hours, recovery actions well outside written and practiced procedures; fire put under control also in 3.5 hours, fire extinguished 2.5 hours later.
Narora 1	PHWR	3/31/93	Turbine blade failure, H_2 explosion and fire, large turbine building fire; fire propagated along cable trays, smoke forced abandonment of MCR (shared between units) for 13 hours, fire caused loss of power to Unit 1 shutdown panel but not Unit 2 shutdown panel; major part of fire put out in 1.5 hours, fire fully extinguished 7.5 hours later; SBO 10 minutes into event, diesel-driven fire pumps used to feed steam generators, tripped by apparent common-cause (but not fire-related) failure 3.5 hours later, one pump restarted 1.75 hours later; operators "flying blind" for 4.5 hours (until staff entered containment to read instruments); EDG started and loaded 5.5 hours into event but shutdown cooling pump not energized until 17 hours – this led to declared end of SBO conditions.

Tables 4 thru 6 represent a very small fraction of the fire events that have occurred. For the U.S. alone, the EPRI Fire Events Database includes reviewed records for nearly 1700 fire events occurring over the period 1990 through 2009 [30]. However, the vast majority of these events have posed minor challenges to nuclear safety and are not addressed in our current, high-level analysis. (An integrated review of these events similar in spirit to that done in NUREG/CR-6738 would likely be useful in an analysis of fire PRA modelling of intermediate, pre-core damage plant states.)

4.2.3 Comparison of Fire PRA and observed scenarios

Qualitatively comparing the U.S. and international precursor descriptions with the fire PRA results, it appears that the fire PRAs are doing reasonably well with respect to our first point of comparison: most of the important scenarios identified by the fire PRAs appear to have a basis in operating experience.

The one major potential concern arises from the high risk importance given to yard fires by the three SPAR-AHZ models (and the associated licensee NFPA 805 models). Yard fires (including large station transformer fires, have been reported – a recent review identifies 50 relevant events in the 1985-2012 time period – but none have been assessed to be significant precursors. At this point, we do not know if this concern applies to a broader set of current fire PRAs.

A somewhat lesser potential concern is revealed by a more quantitative look at the intermediate results of the three SPAR-AHZ models. As indicated earlier, these results suggest a high rate of fire-induced reactor trips (on the order of 0.1/ry) and fire-induced LOOPs (on the order of 0.01/ry).

Reviewing the LERs for 1980-2012, it appears that the U.S. average rates (based on around 80 fire-related trips and 7 fire-related LOOP events in that time period) are on the order of 0.03/ry and 2E-3/ry, respectively. At this point, we do not know if this apparent conservatism applies to a broader set of current fire PRAs. Also, as discussed earlier in this paper, conservatism in estimated intermediate state frequencies does not necessarily imply conservatism in CDF estimates. However, order-of-magnitude mismatches between the model estimates and empirical experience can erode confidence in the models.

Regarding our second point of comparison, it appears that most of the events listed in Tables 4 thru 6 represent, at a high level, scenarios involving fire sources and induced transients typically included in fire PRAs. (The 1989 Oconee fire, which led to an overcooling transient with a potential challenge to reactor pressure vessel integrity, may be an exception.[19]) However, it also appears that the U.S. precursors have involved a number of features not addressed in current fire PRAs:

- multiple fires (e.g., the 2010 Robinson event);
- multiple hazards (e.g., the 1984 Rancho Seco fire where debris from the hydrogen explosion appears to have been the principal cause of damage);
- fires as consequences rather than initiators of a scenario (e.g., the second fire during the 2010 Robinson event).

These observations echo a number of points made by NUREG/CR-6738 in its detailed review of 30 notable fire events (including the Browns Ferry fire and the non-U.S., severely challenging events summarized in Table 6). NUREG/CR-6738, which was specifically intended to identify potential areas for fire PRA technology improvement based on lessons from operational events, states that “the overall structure of a typical fire PRA can appropriately capture the dominant factors involved in a fire incident” but also notes several modeling challenges. These include the treatment of:

- factors underlying long-duration fires (including delays in initiating fire fighting, use of ineffective media in initial attacks, initial fire severity, and fire inaccessibility);
- the effect of smoke propagation on fire fighting and operations;
- personnel actions taken to facilitate fire fighting (including equipment de-energization);
- turbine building fires and fires in non-safety areas;
- fire-induced spurious operation of equipment;
- the effects of fire-induced failures of major structures;
- multiple fires (including multiple fires caused by the same root cause and secondary fires); and
- multiple hazards (including explosions, missiles, and flooding).

NUREG/CR-6738 also indicates that the lack of credit for non-proceduralized operator actions in typical fire PRAs is a source of conservatism, but does not emphasize this point.

[19] *Following normal PRA modeling practices, the 2002 Watts Bar fire, in which the plant’s response to an offsite fire degraded the plant’s capability to tackle onsite fires, is outside the scope of the fire PRA (which is limited to internal fires). However, the authors are unaware of any PRA that has explicitly modeled such a situation in its analysis of LOOP or external fire events.*

Some, but not all of these challenges are being addressed in more recent fire PRAs and ongoing research and development activities.

4.3 Fire PRA analysis technology

The preceding section focuses on the results of current fire PRAs. This section briefly discusses the status of methods, models, tools, and data available for fire PRA.

As pointed out by numerous papers (see Ref. 27 for a recent overview), the basic fire PRA framework and approach remains largely as described by Apostolakis et al. [25, 44] and the PRA Procedures Guide, NUREG/CR-2300 [26]. However, in the years since the initial applications of this methodology (e.g., the early 1980's Indian Point PRA), considerable work has been performed to improve the realism of specific modeling elements. In the late 1990's, the NRC's Office of Nuclear Regulatory Research (RES) initiated a fire PRA research program whose efforts were guided by a structured identification and evaluation of potential problem areas [46]. Using the results of that program and parallel industry activities, RES and EPRI jointly developed NUREG/CR-6850 (EPRI 1011989) [8] and Supplement 1 to that document [9]. These documents provide the principal technical guidance available for current U.S. fire PRAs.

Recent evaluations of the status of fire PRA technology based on NFPA 805 applications have been provided in 2011 by Stetkar et al. [12] and Gallucci [14], and in 2013 by the Nuclear Energy Institute (NEI) [13]. The fire PRA technical issues identified by NEI include:

- the probability of fire-induced short circuits ("hot shorts");
- the duration of fire-induced hot shorts in direct current (DC) circuits;
- the effectiveness of incipient detection systems; and
- the frequency-magnitude relationship for the heat release rates associated with actual plant fires.

Work is ongoing by RES and industry to address this list of issues, which is shorter than earlier lists (e.g., see [12, 14]). Thus, it seems clear that progress towards improved realism is being made. However, it should be recognized that a number of the important (but admittedly extremely difficult) issues identified in NUREG/CR-6738, namely multiple fires, multiple hazards, and non-proceduralized actions, are not yet being addressed.

5. Summary Observations and Questions

Our discussion on fire PRA maturity is heavily influenced by one expert's views on the characteristics of a mature technical field. Our discussion on fire PRA realism relies heavily on: (1) sparse operational data (the number of fire-related precursor events is small; the number of challenging events is even smaller) and the key assumption of exchangeability; (2) summary information provided in recent risk-informed LARs; and (3) detailed information from a very small set of SPAR-AHZ fire models. Furthermore, our review of issues raised by other authors (e.g., potential reasons for differences between fire PRA technology and applications [12], cultural drivers for potential conservatisms [13]) is ongoing. We therefore refrain from drawing definitive conclusions, offering only a number of summary observations from our work to date.

- As a field, fire PRA has been judged sufficiently mature to support important regulatory decisions, but lags internal events analyses in a number of key indicators of technical maturity.

- Comparisons of precursor- and fire PRA-based estimates of the likelihood of a fire-induced core damage event in the U.S. do not show a large numerical difference.
- Comparisons of past and current fire PRAs show that the estimated relative contribution of fire to total CDF has increased significantly.[20]
- With one potentially major exception, most of the important scenarios identified by a number of current fire PRAs appear to have a basis in operating experience. The exception involves yard fires: the high importance ascribed to these fires by the PRAs does not seem to be consistent with actual fire events.
- A number of past fire-related precursor events (U.S. and international) exhibit some important characteristics not reflected in current fire PRAs. Treatment of some of these characteristics (i.e., multiple fires, multiple hazards) would likely increase fire CDF estimates. Treatment of others (particularly non-proceduralized operator recovery actions) would likely decrease fire CDF estimates.
- Recent and ongoing research and development efforts are reducing concerns about the realism of fire PRA's treatment of a number of key issues.
- NUREG/CR-6738 [28] and the report by Stetkar et al. [12] are extremely valuable resources, the former for its in-depth, fire-PRA oriented review of notable fire events, and the latter for its coverage of current issues with the practice of fire PRA and the use of fire PRA results.

Our review also suggests a number of questions whose answers will likely be useful in planning future activities.

- Does the issue of fire PRA maturity warrant additional activity beyond what's being done to improve realism?
- Have there been any recent international important, fire-related precursor events? Do these show the same characteristics as exhibited by U.S. events?
- How do the quantitative and qualitative results of international fire PRAs compare with U.S. results?
- Over the years, have there been any major changes in international perceptions regarding the key contributors to fire risk?
- Does the international PRA community have concerns regarding the realism of fire PRA? If so, do these concerns affect the use of fire PRA results in practical applications?
- What are the key outstanding technical issues in international fire PRAs? Do these need to be resolved to alter the use of fire PRA results?

Acknowledgements

A portion of the technical basis work underlying this paper was performed while one of the authors was serving on rotation in the Office of Commissioner George Apostolakis. The Commissioner's guidance and comments, and the comments from N. Gilles and B. Sosa of his staff, are gratefully appreciated.

[20] *We emphasize that this is an observation. As discussed in Section 4.1.3, the implications may be troubling from a realism perspective and deserving of further investigation.*

The authors also gratefully acknowledge comments from K. Coyne, the help provided by C. Hunter, D. Marksberry, and S. Khericha, and especially the special efforts of J. Baker and E. Paté-Cornell in obtaining material used in this paper.

References

[1] Kolb, G.J., et al., "Review and Evaluation of the Indian Point Probabilistic Safety Study," *NUREG/CR-2934*, 1982.

[2] U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," *NUREG-1150*, 1990.

[3] Payne, A.C., "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)," *NUREG/CR-4832*, Vol. 1, 1992.[4]Rubin, A., et al., "The U.S. Nuclear Regulatory Commission's Review of Licensees' Individual Plant Examination of External Events (IPEEE) Submittals: Fire Analyses," *Proceedings of PSAM 5, International Conference on Probabilistic Safety Assessment and Management*, Osaka, Japan, November 27-December 1, 2000.

[5] Organization for Economic Cooperation and Development, "Fire probabilistic safety assessment for nuclear power plants," *CSNI Technical Opinion Paper No. 1*, Nuclear Energy Agency, Paris, France, 2002. (Available at www.oecd-nea.org/nsd/reports/nea3948-fire-seismic.pdf)

[6] Organization for Economic Cooperation and Development, "Use and Development of Probabilistic Safety Assessment: An Overview of the Situation at the End of 2010," *NEA/CSNI (2012)11*, Nuclear Energy Agency, Paris, France, 2012. (Available at www.oecd-nea.org/nsd/docs/2012/csni-r2012-11.pdf)

[7] National Fire Protection Association, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," *NFPA 805, 2001 Edition*, Quincy, MA, 2001. (Available through the NFPA Online Catalog at www.nfpa.org)

[8] Electric Power Research Institute and U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," *EPRI 1011989 and NUREG/CR-6850*, Electric Power Research Institute (EPRI), Palo Alto, CA and U.S. Nuclear Regulatory Commission, Washington, DC, 2005.

[9] Electric Power Research Institute and U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, "Fire Probabilistic Risk Assessment Methods Enhancements: Supplement 1 to NUREG/CR-6850 and EPRI 1011989," *EPRI 1019259 and NUREG/CR-6850 Supplement 1*, Electric Power Research Institute (EPRI), Palo Alto, CA and U.S. Nuclear Regulatory Commission, Washington, DC, 2009.

[10] U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," *NUREG-1635*, Vol. 1, 1998.

[11] Nuclear Energy Institute, "Insights from the Application of Current Fire PRA Methods for NFPA-805," attachment to letter from B. Bradley, Nuclear Energy Institute to M. Cunningham, U.S. Nuclear Regulatory Commission, January 23, 2008. (Available from the NRC's Agencywide Documents Access and Management System – ADAMS – Accession Number ML080240244)

[12] Stetkar, J.W., W.J. Shack, and H.P. Nourbakhsh, “The Current State of Transition to Risk-Informed Performance-Based Fire Protection Programs,” U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, February 2011. (ADAMS Accession Number ML110430035)

[13] Pietrangelo, A.R., Nuclear Energy Institute, “Industry support and use of PRA and risk-informed regulation,” letter to A.M. Macfarlane, Chairman, U.S. Nuclear Regulatory Commission, December 19, 2013. (ADAMS Accession Number ML13354B997)

[14] Gallucci, R.H.V., “How immature and overly conservative is fire PRA? (A comparison of early vs. contemporary fire PRAs and methods),” *Proceedings of ANS PSA 2011 International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Wilmington, NC, March 13-17, 2011.

[15] U.S. Nuclear Regulatory Commission, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” *Federal Register*, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

[16] Diaz, N., “Realism and Conservatism,” Speech at 2003 Nuclear Safety Research Conference, S-03-023, October 20, 2003. (ADAMS Accession Number ML032940250)

[17] U.S. Nuclear Regulatory Commission, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” *NUREG-1855*, 2009.

[18] Lewis, H., et al., “Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission,” *NUREG/CR-0400*, 1978.

[19] Budnitz, R.J., “Current status of methodologies for seismic probabilistic safety analysis,” *Reliability Engineering and System Safety*, Vol. 62, 71-88(1998).

[20] Organization for Economic Cooperation and Development, “Seismic probabilistic safety assessment for nuclear facilities,” *CSNI Technical Opinion Paper No. 2*, Nuclear Energy Agency, Paris, France, 2002. (Available at www.oecd-nea.org/nsd/reports/nea3948-fire-seismic.pdf)

[21] Cornell, C.A., “Structural safety: some historical evidence that it is a healthy adolescent,” *Proceedings of Third International Conference on Structural Safety and Reliability (ICOSSAR ‘81)*, Trondheim, Norway, June 23-25, 1981.

[22] U.S. Nuclear Regulatory Commission, “In the Matter of Docket Nos. 50-247-SP and 50-286-SP,” CLI-85-6, 21 NRC 1043 (1985). In *Nuclear Regulatory Commission Issuances: Opinions and Decisions of the Nuclear Regulatory Commission, with Selected Orders*, Vol. 21, Book II of II, May 1, 1985 – June 30, 1985. (Available from U.S. Government Printing Office, Washington, D.C.)

[23] U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board, “In the Matter of Docket Nos. 50-247-SP and 50-286-SP (ASLBP No. 81-466-03-SP),” LBP-83-68, 18 NRC 811 (1983). In *Nuclear Regulatory Commission Issuances: Opinions and Decisions of the Nuclear Regulatory Commission, with Selected Orders*, Vol. 18, July 1, 1983 – December 31, 1983. (Available from U.S. Government Printing Office, Washington, D.C.)

[24] Hamzehee, H., “Status of risk-informed regulatory reviews in NRR and associated challenges,” Regulatory Information Conference (RIC) 2014, March 11-13, 2014.

[25] Apostolakis, G., M. Kazarians, and D.C. Bley, “Methodology for assessing the risk from cable fires,” *Nuclear Safety*, **23**, 391-407(1982).

[26] American Nuclear Society and the Institute of Electrical and Electronics Engineers, "PRA Procedures Guide," *NUREG/CR-2300*, 1983.

[27] Siu, N. Melly, S.P. Nowlen, and M. Kazarians, "Fire Risk Analysis for Nuclear Power Plants," draft submitted for publication in the *Society for Fire Protection Engineers' Handbook of Fire Protection Engineering*, 2012. (ADAMS Accession Number ML14084A314)

[28] Nowlen, S.P., M. Kazarians, and F.Wyant, "Risk Methods Insights Gained From Fire Incidents," *NUREG/CR-6738*, 2001.

[29] U.S. Nuclear Regulatory Commission, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," SECY-13-0107, October 4, 2013. (ADAMS Accession Number ML13232A062)

[30] Baranowsky, P.W. and J.W. Facemire, "The Updated Fire Events Database: Description of Content and Fire Event Classification Guidance," *TR1025284*, Electric Power Research Institute, Palo Alto, CA, 2013.

[31] Lelieveld, J., D. Kunkel, and M. G. Lawrence, "Global risk of radioactive fallout after major nuclear reactor accidents," *Atmos. Chem. Phys.*, **12**, 4245–4258(2012).

[32] Kaiser, J.C., "Empirical risk analysis of severe reactor accidents in nuclear power plants after Fukushima," *Science and Technology of Nuclear Installations*, doi:10.1155/2012/384987, 2012.

[33] Gallucci, R., "'What—me worry?' 'Why so serious?': A personal view on the Fukushima nuclear reactor accidents," *Risk Analysis*, doi: 10.1111/j.1539-6924.2011.01780, 2012.

[34] Apostolakis, G., "Global statistics vs. PRA results: which should we use?" Regulatory Information Conference (RIC) 2014, March 11-13, 2014. (Viewgraphs available from www.nrc.gov/about-nrc/organization/commission/comm-george-apostolakis/testimony-speeches.html#speeches)

[35] Garrick, B.J., "Lessons learned from 21 nuclear plant probabilistic risk assessments," *Nuclear Technology*, **84**, No. 3, 319-330(1989).

[36] Gallucci, R.H.V., "Predicting fire-induced core damage frequencies – a simple 'sanity check'," *Transactions of 2006 American Nuclear Society Annual Meeting*, Vol. 94, Reno, NV, June 2006.

[37] U.S. Nuclear Regulatory Commission, "The Browns Ferry Fire Nuclear Plant Fire of 1975 Knowledge Management Digest," *NUREG/KM-0002*, 2013.

[38] U.S. Nuclear Regulatory Commission, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," SECY-10-0125, September 29, 2010. (ADAMS Accession Number ML102100313)

[39] U.S. Nuclear Regulatory Commission, "Status of Accident Sequence Precursor and SPAR Model Development Programs," SECY-02-0041, March 8, 2002. (ADAMS Accession Number ML020420319)

[40] U.S. Nuclear Regulatory Commission, "Status of the Accident Sequence Precursor (ASP) Program and the Development of Standardized Plant Analysis Risk (SPAR) Models," SECY-05-0192, October 24, 2005. (ADAMS Accession Number ML052700542)

[41] Atwood, C.L., et al., "Handbook of Parameter Estimation for Probabilistic Risk Assessment," *NUREG/CR-6823*, 2003.

[42] Lambright, J.A., et al., “Analysis of the LaSalle 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Internal Fire Analysis,” *NUREG/CR-4832, Vol. 9,* 1993.

[43] U.S. Nuclear Regulatory Commission, “Reliability and Probabilistic Risk Assessment - June 22, 2001,” Official Transcript of Proceedings, Meeting of Advisory Committee on Reactor Safeguards Subcommittee on Reliability and Probabilistic Risk Assessment, June 22, 2001. (Available at www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2001/pr010622.html)

[44] Kazarians, M., N. Siu, and G. Apostolakis, “Fire risk analysis for nuclear power plants: methodological developments and applications, *Risk Analysis*, **5**, 33-51 (1985).

[45] Sancaktar, et al., “Incorporation of all hazard categories into U.S. NRC PRA models,” *Proceedings of International Workshop on PSA for Natural Hazards Including Earthquakes*, Prague, Czech Republic, June 17-19, 2013 (in publication).

[46] Siu, N., J.T. Chen, and E. Chelliah, “Research needs in fire risk assessment,” Proceedings of 25th U.S. Nuclear Regulatory Commission Water Reactor Safety Information Meeting, *NUREG/CP-0162, Vol. 2*, 1997.

Fire Ignition Frequency Estimation: Component-based versus Room-based Approaches

Roman Grygoruk, Hartmut Schmaltz, Heiko Kollasko
AREVA GmbH, Germany

Abstract

Different approaches are described for the assessment of fire ignition frequencies in the frame of a Fire PRA. For the most of German Fire PSA the room-based, known also as a "Berry" method 0, is applied as proposed in 0. Applying the Berry method the relative fire ignition frequency of each fire compartment is determined according to aspects which can influence the frequency (e.g. distribution of combustible material in the fire compartment, ignition temperature of the combustible material, ignition sources). The absolute fire ignition frequency of each fire compartment is derived using the relative probability of the fire within the analyzed compartment and the absolute fire ignition frequency for the building containing the given compartment. The absolute fire ignition frequency for the building is taken from plant specific operational experience or generic data sources. Plant walk downs are to be performed to support collection of data on fire loads and ignition sources.

In recent decade the method as described in 0 has been established for application in Fire PSA. This method bases the assessment of fire frequencies on equipment types as ignition source. A plant-wide specific fire ignition frequency corresponding to the total frequency of fires caused by each specific equipment type of ignition sources is available based on operating experience of US NPPs. The ignition sources comprise components as well as hydrogen fires, transient fires and fires caused by welding and cutting. For determining the fire ignition frequency for a specific kind of components located within analyzed fire compartment, the total fire ignition frequency of a given ignition source is weighted by the proportion of the number of components that are physically located in the analyzed fire compartment to the total number of similar components plant-wide. Then, the ignition frequency of the fire within the fire compartment derives from the sum of the calculated fire ignition frequencies for the ignition sources that are located within the fire compartment.

AREVA GmbH has performed the Fire PSA for various NPPs. In the frame of these analyses, AREVA GmbH has gained experience with both approaches. This paper presents the comparison of two approaches for fire ignition frequency estimation.

1. Short Description of Approaches

1.1 Estimation of fire ignition frequencies using room-based method

Using the room-based, also known as "Berry" method, the relative fire frequency of each fire compartment is determined according to aspects which can influence the fire ignition frequency (e. g. distribution of combustible material in the fire compartment, ignition temperature of the combustible material, ignition sources). The absolute fire frequencies of each fire compartment are derived using the relative fire probability of the fire compartment and the absolute fire frequency of the buildings. The absolute fire ignition frequency for the building is taken from plant specific operational experience or generic data sources e.g. from 0.

The following procedure is to be used

1. Step 1: Each room is to be examined during plant walk down taking into account additional information about component location. The result to be documented separately.
2. Step 2: Based on the results of step 1 the Berry parameter A_1, A_2, A_3, B, C_1, C_2 and F to be determined using the check list given as example in Table 1.
3. Step 3: In this step the calculation of conditional probabilities is performed. The conditional probabilities that a fire which starts in a building "a" starts in a specific room "k" are calculated using the following equation:

   $$P_k^a = A \cdot B \cdot (1-(C_1 \cdot C_2)) \cdot (1-F)$$

   where

   - Parameter A –
     characterizes ignition sources and is calculated by the following equation:

     $$A = 1-(1-A_1)\cdot(1-A_2)\cdot(1-A_3)$$

     where
     - Parameter A_1 – characterizes the duration of personal in room
     - Parameter A_2 – characterizes the amount of mechanical components
     - Parameter A_3 – characterizes the amount of electrical components
   - Parameter B – ignition probability (depends on the flash point of the combustible material)
   - Parameter C_1 – characterizes the presence of people in rooms to discover a developing fire
   - Parameter C_2 – characterizes the possibility that the fire will be extinguished (depends on the flash point of the combustible material)
   - Parameter F – characterizes the possibility of self-extinguishing (depends on the distribution of the combustion material in the fire compartment)
4. Step 4: Based on the calculated probabilities per each room P_k^a the frequency of a fire in room "k" F_k^a is to be determined:

   $$F_k^a = F_a \cdot \frac{P_k^a}{\sum_{i=1}^{n} P_i^a}$$

   where

   F_a – the fire frequency in building "a", and

   $\sum_{i=1}^{n} P_i^a$ - the sum of the conditional probabilities of a fire by all rooms in building "a".
5. Step 5: Finally, the fire frequencies calculated for each room are summed up for each defined fire compartment.

Table 1. **Example of the room check list**

Room-no.: ______________

Case 1	**A_1**	**C_1**		**Comments:**
Personnel are present in the room				
- all the time	0,7	0,99	☐	....
- most part of the time	0,7	0,95	☐	....
- a third of the time	0,3	0,90	☐	....
- during the rounds	0,2	0,1	☐	....
- seldom	0,1	0	☐	....
Case 2	**A_2**			**Comments:**
The amount of mechanical equipment				
large	0,5		☐	....
average	0,3		☐	....
small	0,1		☐	....
Case 3	**A_3**			**Comments:**
The amount of electrical equipment				
- large	0,3		☐	....
- average	0,1		☐	....
- small	0,05		☐	....
Case 4	**B**	**C_2**		**Comments:**
Comb. material always present. Easy to ignite, flash point ≤ 20°C.	1,0	0,5	☐	....
As above, but flash point between 20°C and 250°C.	0,1	0,9	☐	....
As above, but flash point above 250°C.	0,01	0,99	☐	....
Others	0,01	0,99	☐	....
Case 5	**F**			**Comments:**
Comb. material present in				
- the entire room	0,02		☐	....
- the major part of the room	0,20		☐	....
- the half of the room	0,5		☐	....
- the minor part of the room	0,9		☐	....
Comb. material are normally not present	0,95		☐	....

1.2 Fire ignition frequencies derived from ignition sources – component-based method

The method to assess fire frequencies using the equipment type (ignition source) approach as described in detail in 0 is used more and more for Fire PSA. A plant-wide specific ignition frequency corresponding to the total frequency of fires caused by each specific type of ignition source is available in 0 and bases on operating experience of US NPPs. Each record in the EPRI Fire Events Database (FEDB), ref. 0, evaluated and included accordingly in the frequency model is assigned to a bin. The ignition sources – bins – comprise components and others, e.g. hydrogen fires or fires from welding and cutting. This number has to be weighted by the proportion of the total number of components that are present in that specific fire area. Thus, for each fire area J and for each type of component IS an ignition frequency for the component type has to be determined by multiplying the ignition frequency for the component type λ_{IS} from 0 by the fraction of the component type that is physically located within in the fire area $W_{IS,J}$.

$$\lambda_{IS,J} = \lambda_{IS}\, W_L\, W_{IS,J}$$

where

λ_{IS} Plant-level fire frequency associated with ignition source IS

W_L Location weighting factor associated with the ignition source

$W_{IS,J,L}$ Ignition source weighting factor reflecting the quantity of the ignition source type present in compartment J of location L.

The ignition frequency of the fire area J is the sum of the adjusted ignition frequencies for the component types that are located within the fire area. The main sources which can be used to evaluate the number of components per location are the electrical consumer list that lists all components that have a power supply and cable list.

The component-based analysis model is based on the following assumptions:

- Fire ignition frequencies remain constant over time;
- Among the plants, total ignition frequency is the same for the same equipment type, regardless of differences in the quantity and characteristics of the equipment type that may exist among the plants;
- Within each plant, the likelihood of fire ignition is the same across an equipment type. For example, pumps are assumed to have the same fire ignition frequency regardless of size, usage level, working environment, etc.

The procedure to estimate fire ignition frequency using component-based method is organized around the following steps:

Step 1: Mapping plant ignition sources to generic sources.

The purpose of this step is to map all plant components that can initiate a fire (e.g., electrical equipment) to a corresponding bin (see Table 2) as defined in 0. For each fire compartment/room, a review of all ignition sources should be conducted to verify that every ignition source can be mapped to one of the relevant bins.

Table 2. **Fire ignition sources from 0**

Bin	Location	Ignition Source (Equipment Type)	Ignition Source Category
1	Battery Room	Batteries	Countable items
2	Containment (PWR)	Reactor Coolant Pump	Countable items
3	Containment (PWR)	Transients and Hotwork	Transients
4	Control Room	Main Control Board	Countable items
5	Control/Auxiliary/Reactor Building	Cable fires caused by welding and cutting	Transients
6	Control/Auxiliary/Reactor Building	Transient fires caused by welding and cutting	Transients
7	Control/Auxiliary/Reactor Building	Transients	Transients
8	Diesel Generator Room	Diesel Generators	Countable items
9	Plant-Wide	Air Compressors	Countable items
10	Plant-Wide	Battery Chargers	Countable items
11	Plant-Wide	Cable fires caused by welding and cutting	Transients
12	Plant-Wide	Cable Run (Self-ignited cable fires)	Countable items
13	Plant-Wide	Dryers	Countable items
14	Plant-Wide	Electric Motors	Countable items
15	Plant-Wide	Electrical Cabinets	Countable items
16	Plant-Wide	High Energy Arcing Faults	Countable items
17	Plant-Wide	Hydrogen Tanks	Countable items
18	Plant-Wide	Junction Boxes	Countable items
19	Plant-Wide	Misc. Hydrogen Fires	Large Systems
20	Plant-Wide	Off-gas/H2 Recombiner (BWR)	Large Systems
21	Plant-Wide	Pumps	Countable items
22	Plant-Wide	RPS MG Sets	Countable items
23	Plant-Wide	Transformers (Dry)	Countable items
24	Plant-Wide	Transient fires caused by welding and cutting	Transients
25	Plant-Wide	Transients	Transients
26	Plant-Wide	Ventilation Subsystems	Countable items

Bin	Location	Ignition Source (Equipment Type)	Ignition Source Category
27	Transformer Yard	Transformer – Catastrophic	Countable items
28	Transformer Yard	Transformer - Non Catastrophic	Countable items
29	Transformer Yard	Yard transformers (Others)	Countable items
30	Turbine Building	Boiler	Countable items
31	Turbine Building	Cable fires caused by welding and cutting	Transients
32	Turbine Building	Main Feedwater Pumps	Countable items
33	Turbine Building	Turbine Generator Excitor	Countable items
34	Turbine Building	Turbine Generator Hydrogen	Large Systems
35	Turbine Building	Turbine Generator Oil	Large Systems
36	Turbine Building	Transient fires caused by welding and cutting	Transients
37	Turbine Building	Transients	Transients

Step 2: Plant fire event data collection and review.

The generic fire frequencies are to be updated using plant-specific fire event data if available. Use of the generic fire frequency data is reasonable if an important condition is met: there are no unusual fire occurrence patterns in the plant.

Step 3: Plant specific updates of generic ignition frequencies.

This step is to be followed for those frequencies that will be based on plant-specific fire event data. After the plant-specific data has been collected and analyzed in Step 2, the generic bin frequencies can be updated using Bayesian approach.

Step 4: Mapping plant-specific locations to generic locations.

This step maps plant-specific locations to generic locations defined in 0. The following set of generic plant locations is used in defining ignition source bins:

- Battery Room,
- Containment (PWR),
- Control Room,
- Control/Auxiliary/Reactor Building,
- Diesel Generator Room,
- Plant-Wide Components,
- Transformer Yard, and
- Turbine Building.

Step 5: Location weighting factors.

The location weighting factor is used to adjust the generic fire frequencies to account for locations and/or equipment shared among the units in multi-unit sites. Thus, location weighting factors, W_L, only apply to multi-unit sites. For single-unit sites, W_L=1.0 is used. However, if it is possible to obtain a separate equipment count for each unit in a multi-unit site, the W_L=1.0 can be applied.

Step 6: Fixed fire ignition source counts.

To establish an ignition source weighting factor, $W_{IS,J}$, per compartment, it is necessary to obtain the total number of items per the equipment type (bins) and number of items physically located in analyzed compartment.

Step 7: Ignition source weighting factors.

Ignition source weighting factor, $W_{IS,J,L}$, is the fraction of ignition source (IS) that is present in compartment J. The $W_{IS,J,L}$ are evaluated for all relevant compartments and for all ignition sources identified in Step 1. The bins listed in Table 2 can be classified in three categories: countable items, transients, and large systems.

The ignition source weighting factor, $W_{IS,J,L}$, for countable items is calculated by dividing the number of each IS in compartment J by the total number in the generic locations obtained in the previous step.

A relative ranking scheme is used for estimating the ignition source weighting factors for ignition frequency bins involving transient combustibles or activities. Occupancy level, storage of flammable materials, and type and frequency of maintenance activities in a compartment are the three most important influencing factors of the likelihood of fire ignition involving a transient combustible or activity. It is recommended to use the following five rating levels:

1. No (0) – Can be used only for those compartments where transients are precluded by design.
2. Low (1) – Reflects minimal level of the factor.
3. Medium (3) – Reflects average level of the factor.
4. High (10) – Reflects the higher-than-average level of the factor.
5. Very high (50) – Reflects the significantly higher-than-average level of the factor (only for “maintenance” influencing factor).

For category “Large Systems” a geometric factor may be used to adjust bin frequency to the specific area of the plant where the components addressed in the bin could be risk-significant. The geometric factor refers i.e. to floor area ratio.

Step 8: Ignition source and compartment fire ignition frequency evaluation

The fire frequency (generic or plant-specific) for each ignition source, $\lambda_{IS,J}$, can now be calculated using the data quantified in the preceding steps with the equation

$$\lambda_{IS,J} = \lambda_{IS} W_L W_{IS,J}$$

Compartment level fire frequency would then be calculated from:

$$\lambda_J = \Sigma \lambda_{IS} W_L W_{IS,J}$$

(Summed over all ignition sources IS in compartment J of location L)

2. Qualitative Comparison of Two Approaches

2.1 General procedure

The *room-based method* starts with building fire ignition frequencies which are then broken down to rooms based on fire loads determined during plant walk-down and recorded in check list (example is presented in Table 1). To be noted, that the method requires that the plant walk-down to be performed by the same analyst (at least for a building) in order to ensure that the evaluation criteria are the same.

The *component-based method* starts with the total plant fire ignition frequency for each fire source group. It is therefore necessary to distribute this total generic frequency among the different plant locations or fire areas where the different types of fire sources may be found.

<u>*Conclusion:*</u> The both methods can be treated as “top-down” – starting with total building/plant fire ignition frequency and finishing by determining the ignition frequency for analyzed room/compartment.

2.2 Consideration of components as ignition sources

Using the *room-based method,* all relevant rooms are compared and their inventory is assessed during a plant walk-down. There is no relevance on component types installed in the room, except distinction between mechanical and electrical equipment (see parameters A2 and A3. Rooms are categorized with respect to the amount of inventory in the room i.e. large, average or small amount of electrical or mechanical equipment located in analyzed room.

The *component-based method* requires that plant wide ignition sources, among others components, are binned, counted then the relevant rooms are evaluated by applying the weighting factors.

<u>*Conclusion:*</u> The component-based method is considered to be more precise with regard to the evaluation of potential ignition sources in the room due to the fact that component types are distinguished more precisely.

2.3 Consideration of transients as ignition sources

The *room-based method* considers only the presence of personnel as factor to determine transient ignition sources (parameter A_1).

For transient fires the *component-based method* considers also the presence of personnel (occupancy level) however, additionally, takes into account such factors as storage of flammable materials, as well as type and frequency of maintenance activities in analyzed compartments.

<u>*Conclusion:*</u> Rooms with little or no combustible equipment/material are analysed with respect to transient fires in both methods. Component-based method is considered to be more precise in counting of transient ignition sources as additionally to the presence of personnel, the storage of flammable material and the type and frequency of activities in the room are taken into account.

2.4 Available data

For the *room-based method* the fire ignition frequency for the related buildings have to be based on plant specific operational experience rather than generic data sources e.g. from 0.

The *component-based method* can be applied using generic data from 0 unless the plan-specific data are available. If plant specific data are available generic data can be updated by using a Bayesian approach.

Conclusion: Generally, the two methods are similar in applying the data - both can use generic data and can be updated by plant-specific data. However, the component-based method uses component specific data for ignition frequency estimation which does not depend on plant type, thus, allows more flexibility in applying the generic data to different plants.

2.5 Fire loads distribution

For the *room-based method* performance of a plant walk-down is generally needed to determine fire loads distribution in the rooms (parameter F) there is, generally, need for plant walk-down.

For fire ignition frequency estimation of the *component-based method* the determination of fire loads distribution within the rooms is not essential, thus, there is no direct need for a plant walk-down.

Conclusion: The fact that plant walk-downs are not necessarily needed allows to apply the component-based method for fire ignition frequency estimation for fire PRA of new build NNPs during design process.

2.6 Parameters comparison

The parameters determined for the fire ignition frequency estimation of the *room-based method* associated to applicable factors of the *component-based method* are presented in Table 3.

Table 3. **Comparison of parameters**

Room-based method	***Component-based method***	***Conclusion***
Parameter A_1 – Duration of personal in room	Determined by occupancy influence factor for transient fires (bins 3, 5, 6, 7, 11, 24, 25, 31, 36 and 37)	The both approaches are similar in consideration of personal presence (rooms occupancy) as influence factor for fire ignition frequency estimation
Parameter A_2 – Amount of mechanical components	Determined by weighting factor for bins 2, 8, 9, 13, 17, 20, 21, 26, 30, 32	For component-based method mechanical equipment is subdivided into equipment types which are physically counted for analyzed rooms while the room-based method considers only three categories of rooms with respect of the amount of inventory. Thus, the component-based method is more detailed in equipment consideration.
Parameter A_3 – Amount of electrical components	Determined by weighting factor for bins 1, 4, 10, 12, 14, 15, 18, 22, 23, 27, 28, 33	For component-based method electrical equipment is subdivided into equipment types which are physically counted for analyzed rooms while the room-based method considers only three categories of rooms with respect of the amount of inventory. Thus, the component-based method is more detailed in equipment consideration.

Room-based method	***Component-based method***	***Conclusion***
Parameter B – Combustion succeeds	Determined by presents of different ignition sources (bins)	The component-based approach explicitly considers equipment as combustibles with their combustions characteristics. For transient fires parameter B in room-based method can be compared with storage influence factor in component-based approach. Thus, the two approaches are similar.
Parameter C_1 – Combustions discovered	Not a part of fire ignition frequency calculation. Considered separately for detection and suppression analysis	Not comparable
Parameter C_2 – Combustions suppressed	Not a part of fire ignition frequency calculation. Considered separately for detection and suppression analysis	Not comparable
Parameter F – Fuel exhausted (self-extinguishing)	Not a part of fire ignition frequency calculation. Considered for detailed fire modelling. Determined by severity factor	Not comparable

3. Summary

The comparison of room-based and component-based approaches to estimate fire ignition frequency for Fire PSA shows that the both approaches are basically similar. Both methods are "top-down" approaches; both are based on relative contribution of fire ignition frequency for each analyzed compartment to the whole building/plant; both characterize rooms by the factors which influence fire ignition frequency (e.g. occupancy and storage levels, amount of electrical and mechanical components etc.). Anyway, the differences exist in both the level of details as well as in single steps implementation.

The component-based method, which counts physical ignition sources (equipment types), is considered to be more detailed and precise for the fire ignition frequency estimation of the analyzed compartments compared to the room-based method which roughly estimates the relative amount of mechanical or electrical components between all analyzed rooms of one building. Therefore, applying room-based approach, the building fire ignition frequency is relatively distributed between rooms, so that, the room specific differences are not fully accounted. Contrariwise, due to direct equipment counting, the component-based approach allows to account for more room specific differences resulting e.g. in considerably higher fire ignition frequency for the rooms with high fire loads (i.e. rooms containing lots of electrical cabinets – switchgear rooms and/or I&C rooms).

The room-based method considers the possibility to detect and suppress an ignition on its incipient stage as well as self-extinguishing of combustion (fire severity). This is not the case for the component-based method where such possibilities are analyzed as separated tasks in detection and suppression analysis and detailed fire modelling.

Considering the results of the comparison the following can be concluded:

1. The component-based approach is more generic. Thus it can be applied for different plant types, including new build plants. Therefore, the component-based approach is more suitable for estimation of fire ignition frequency in Fire PSA during plant design process;
2. The room-based approach already includes such factors as detection, suppression and self-extinguishing of combustion while component-based approach requires to consider these factors separately and more in detail making it more precise, however, requiring more effort.
3. For existing NPP, the room-based approach, which is mostly based on inputs from plant walk-down, is easier to implement and requires less effort to estimate fire ignition frequency with acceptable accuracy level.

References

[1] EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Plant Facilities
NUREG/CR-6850 / EPRI 1011989
Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD:
September 2005

[2] Berry, D. L.; Minor, E. E.
Nuclear Power Plant Fire Protection - Fire-Hazards Analysis
(Subsystem Study Task 4)
NUREG/CR-0654
SAND79-0324, RP.
Sandia Laboratories Albuquerque, New Mexico, September 1979

[3] Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke
Bundesamt für Strahlenschutz
BfS-SCHR-38/05, ISSN 0937-4469, ISBN 3-86509-415-5
August 2005

[4] Methoden zur Probabilistischen Sicherheitsanalyse für Kernkraftwerke
Bundesamt für Strahlenschutz
BfS-SCHR-37/05, ISSN 0937-4469, ISBN 3-86509-414-7
August 2005

[5] Fire Event Database and Generic Ignition Frequency Model for U.S. Nuclear Power Plants.
EPRI, Palo Alto, CA: 2001. 1003111.

Probabilistic set of filter criteria in the frame of Fire PSA

Florian Berchtold[1]*, Burkhard Forell[2], Ulrich Krause[3]
[1]BAM Federal Institute for Materials Research and Testing, Berlin, Germany
[2]Gesellschaft für Anlagen- und Reaktorsicherheit mbH, Cologne, Germany
[3]Otto-von-Guericke University Magdeburg, Magdeburg, Germany
*Corresponding author: florian.berchtold@bam.de

Abstract

Filter criteria in the Frame of Fire PSA identify compartments in a first qualitative analysis for which the contribution to the overall core damage frequency of the NPP is negligible. The aim of the filter criteria is to reduce the number of compartments to be analysed precisely in Fire PSA. One example for filter criteria is the 'fire load criterion'. By the fire load criterion compartments with a fire load density of less than 90 MJ/m² are 'screened out' which means to exclude them from a precise analysis in Fire PSA. Neither the justification of the particular value of 90 MJ/m² is well documented nor does this criterion take into account varying compartment configurations such as ventilation conditions, physical and chemical properties of the fire load as well as compartment characteristics.

A probabilistic set of filter criteria was developed to overcome the restrictions of the fire load criterion. In line with the 'fire load criterion', the probabilistic set of filter criteria assumes that a compartment can be screened out if a fire is not able to cause any damage to other components within the compartment. Therefore, the electrical failure of an electrical cable conservatively represents the damages of all components. It is assumed that the electrical cable failure occurs when the maximum cable temperature exceeds an experimentally determined failure temperature. The maximum cable temperature that can occur in a compartment fire is mainly influenced by the four significant factors: 1. inlet air stream of the mechanical ventilation, 2. the fire growth rate, 3. the compartment floor area and 4. the compartment height. A parameter study revealed how the significant factors affect the maximum cable temperature in fictitious compartment fires. The results of the parameter study are transferred on true Nuclear Power Plant compartments. However, it is not possible to determine precisely the occurrence of an electrical cable failure because of uncertainties in the maximum cable temperature and the failure temperature. The probabilistic set of filter criteria considers these uncertainties and determines the probability of cable failure for true compartments to be screened in Fire PSA. Finally, a compartment can be screened out in Fire PSA if the failure probability exceeds a predefined accepted threshold value for the failure probability. The theoretical application of the methodology is shown at the end of the paper.

1. Introduction

The Probabilistic Fire Safety Analysis (Fire PSA) is mandatory for all operational states of a Nuclear Power Plant (NPP) due to German legal requirements. In the Fire PSA, it is assessed how plant internal fires contribute to the overall core damage frequency of the entire NPP. Since the safety design of a NPP relies on compartmentation of the whole NPP and its buildings, each compartment has to be analysed separately in Fire PSA which is time consuming. Therefore, filter criteria can be applied to identify compartments for which the contribution of plant internal fires to the overall core damage frequency of the NPP is negligible [1]. These compartments can be 'screened out', which means to exclude them from a precise analysis in Fire PSA and thus to reduce the required time for the Fire PSA.

The 'fire load criterion' is a filter criterion of a first qualitative analysis in the Fire PSA [2]. By the fire load criterion compartments with a fire load density of less than 90 MJ/m^2 are screened out. The technical background is the assumption that a fire in a compartment with this fire load density would not be able to cause any damage to other components within the compartment except for the component where the fire originally started. However, neither the justification of the particular value of 90 MJ/m^2 is documented nor takes this criterion into account varying compartment configurations such as ventilation conditions, physical and chemical properties of the fire load as well as compartment characteristics. In addition, the fire load criterion is not always valid e.g. in the case of large compartments with locally concentrated fire loads. For these reasons, the probabilistic set of filter criteria was developed to overcome these restrictions of the fire load criterion.

The structure of the paper is as follows: in Section 2 the methodology of the probabilistic set of filter criteria is described in detail. In Section 3 the fire behaviour in the simulations of the parameter study and the effects of the compartment configuration on cable temperatures are explained. And in Section 4 the methodology is theoretically applied on a compartment configuration and the effects of the random variables on the cable temperature are highlighted. Finally, in Section 5 some conclusions and an outlook for further research are given.

2. Methodology of the Probabilistic Set of Filter Criteria

A new probabilistic set of filter criteria was developed to screen out those compartments from Fire PSA where a fire of one component causes with a certain probability no damage of any other component. Figure 1 illustrates the methodology which principally consists of three parts: 1. the deterministic part, 2. the data transfer and 3. the probabilistic part.

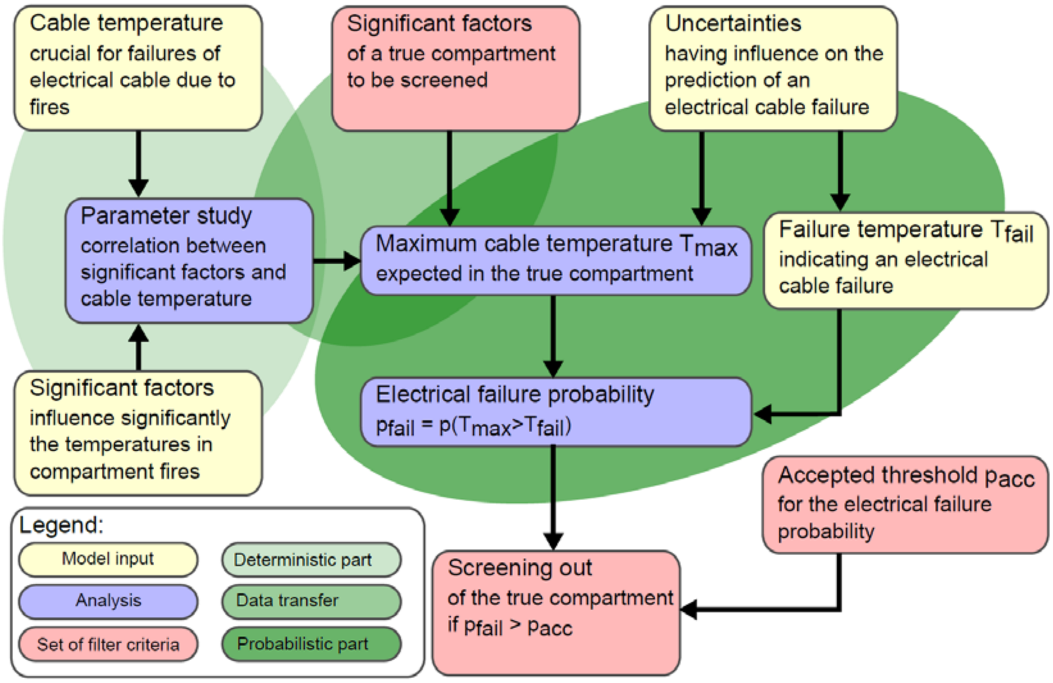


Figure 1. **Scheme of the methodology developed to define the probabilistic set of filter criteria**

This section describes the methodology in detail and is organised as follows: First, the deterministic part of the methodology is described in Subsection 2.1. The deterministic part is based on a parameter study which determines the correlation between the damage of components and four significant factors in fictitious compartments. The electrical failure of an electrical cable conservatively represents the component damage. An electrical cable failure is assumed if the maximum cable temperature exceeds an experimentally determined failure temperature. The four significant factors of a compartment, the mechanical inlet air stream, the fire growth rate, the floor area, and the height, significantly influence the maximum cable temperatures in case of fire. These factors are summarised in the compartment configuration in this methodology. Second, in Subsection 2.2 a method based on linear interpolation is described that transfers the simulation results of the limited number of fictitious compartments in the parameter study to all compartment configurations of true compartments in a NPP. Third, in Subsection 2.3 the probabilistic part of the methodology is explained. The probabilistic part considers different random variables that influence the identification of electrical cable failures. Two probabilistic methods are applied to determine the probability of an electrical cable failure by comparison of the maximum cable temperature with the failure temperature of the cable. A true compartment can be screened out if the failure probability is lower compared to an accepted threshold value.

2.1 Deterministic model of the electrical cable failure and the compartment fire used for the parameter study

In line with the 'fire load criterion', the probabilistic set of filter criteria assumes that a compartment can be screened out if a fire is not able to cause any damage to other components within the compartment. The electrical cable failure conservatively represents the damage of all components of the variety of appliances in a NPP [3]. The term 'electrical cable failure' comprises all possible electrical malfunctions of an electrical cable. The instrumentation and control cable

JE-Y(St)Y 16x2x0.8 (named as 'reference cable') showed in particular early electrical failure during fire tests [4] and was therefore chosen as being representative for all other cables in a NPP.

Results of the Cable Response To Live Fire (CAROLFIRE) Project [5] showed that an electrical cable failure can be predicted by the temperature at the inner side of the cable jacket ('cable temperature') because the insulation of the conductors is crucial for electrical cable failures. It is assumed that the electrical cable failure happens when the cable temperature reaches an experimentally determined failure temperature T_{fail}. The failure temperature of the reference cable was analysed in three experimental studies and is estimated to be in a range of T_{fail} = 240 °C ± 40 °C [3, 4]. To sum up, the maximum cable temperature combined with the failure temperature of the reference cable conservatively represents the damage of components in a compartment.

Different factors influence the maximum heat release rate (HRR) of a compartment fire, which is crucial for the possible maximum cable temperature within the compartment. The maximum HRR in compartments of NPPs is usually limited by the supply of oxygen because the compartments do not have any passive vents such as windows and the mechanical ventilation is usually low [6]. Hence, two main sources of oxygen influence the maximum HRR significantly, namely the inlet air stream of the mechanical ventilation and the compartment volume. A fast development of the fire can also lead to high HRRs in a compartment. Thus, the fire growth rate seems to be important especially at low ventilation rates in small compartments. To summarise, the compartment configuration comprises the four 'significant factors', the inlet air stream of the mechanical ventilation, the fire growth rate, the compartment area, and the compartment height. Hence, the specific compartment configuration of a compartment in a NPP is significant for electrical cable failures due to compartment fires.

A parameter study revealed how the significant factors affect the maximum cable temperature. For this purpose, the significant factors were varied in ranges as summarised in Table 1. The fire growth rate lies between a value well below the slow fire growth rate and a moderate fire growth rate [7].

Table 1. **Significant factors with their minimum and maximum values used in the parameter study**

Significant factor	Minimum value	Maximum value
Inlet air stream	0 m^3/s	4 m^3/s
Fire growth rate	0.000326 kW/s^2	0.0117 kW/s^2
Compartment area	100 m^2	576 m^2
Compartment height	2.6 m	10 m

The Fire Dynamics Simulator 5.5.3 (FDS) [8] was used to build up the fictitious compartment (see Figure 2) and to simulate the fire scenarios. The fictitious compartment was based on several compartments characterised in other fire safety analyses [9–11] as well as on some conservative assumptions. One conservative assumption was the square-formed ground floor in order to minimise the conductive heat loss over the concrete walls. Other conservative assumptions were the positions of the fire source, the inlet air vents and the outlet air vents in comparison to several other arrangements within the compartment.

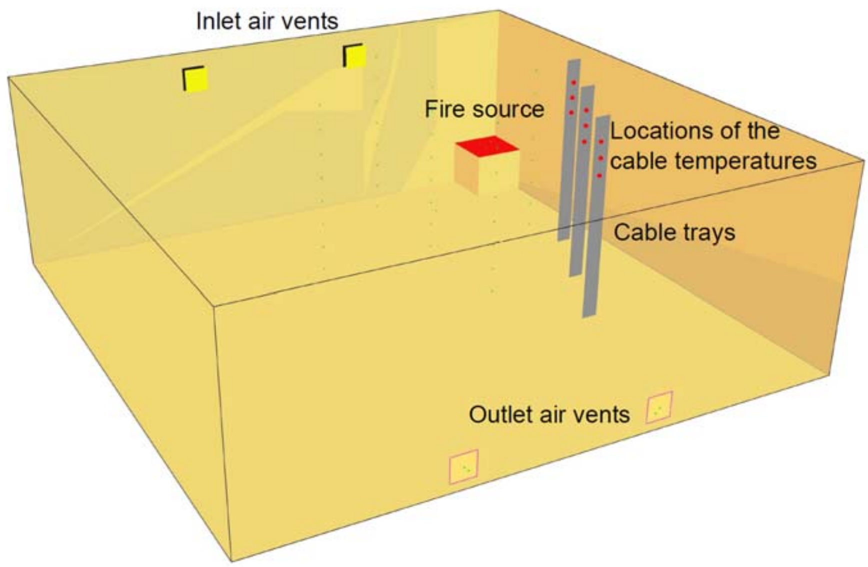


Figure 2. **The conservative arrangement within the fictitious compartment modelled in FDS**

The fire source was located in a corner of the compartment next to the wall that contains the inlet air vents of the mechanical ventilation system. The inlet air vents were placed close to the ceiling and had a constant inlet air stream which was diffused by simple plates. Additionally, the outlet air vents were on the opposite wall of the inlet air vents close to the compartments floor. The outlet air vents were modelled as 'open vents' in FDS. The HRR of the fire source was prescribed by the t-squared approach with the fire growth rate α. The ventilation-limited conditions of the HRR in the compartment are modelled with the two-step reaction with extinction model in FDS [12]. All other models were used according to the default settings of FDS.

The Thermally Induced Electrical Failure (THIEF) model [8] was used to simulate the cable temperature in FDS. The THIEF model is based on the results of the CAROLFIRE Project [5]. The cable temperatures were simulated on three vertical cable trays at three levels of 0.6 m, 1.0 m, and 1.4 m below the ceiling. The three cable trays were positioned in the compartment in a horizontal distance of 3.6 m, 5.6 m, and 7.6 m to the fire source and were modelled as solid steel plates in order to represent the physical presence of the cables.

The accuracy of the fire model was analysed in a grid sensitivity study and a validation study. The grid sensitivity study revealed grid-independent simulation results for grid sizes equal or smaller than 0.2 m. For this reason, the grid size of 0.2 m was used in the parameter study. The validation study was carried out using two experiments from the Fire Propagation in Elementary Multi-room Scenarios (PRISME) program [13] and one experiment conducted in the frame of the International Collaborative Fire Model Program (ICFMP) Benchmarking and Validation Exercise #3 [14]. The compartments were mechanically ventilated and the cable temperatures of several cables, among them the reference cable, were measured in all three experiments. The result of the validation study implied that the fire model underestimates the maximum cable temperature by the factor $\delta \approx -0.2$ in relation to the experimental results. This systematic uncertainty was taken into account in the methodology by equation 1.

$$T_\delta = T / (1 + \delta) \tag{1}$$

where T_δ takes into account the underestimation of the cable temperatures T in FDS and thus is closer to the experimental results. More information on the fire model and the validation process are summarised in [15].

2.2 Application of the parameter study on true compartment configurations

A method was developed to transfer the information of the limited number of fire simulations in fictitious compartments on any compartment configuration of true compartments in NPPs. Basically, the method comprises a linear interpolation of the simulation results along the four dimensions of the significant factors. Figure 3 illustrates the four dimensional linear interpolation.

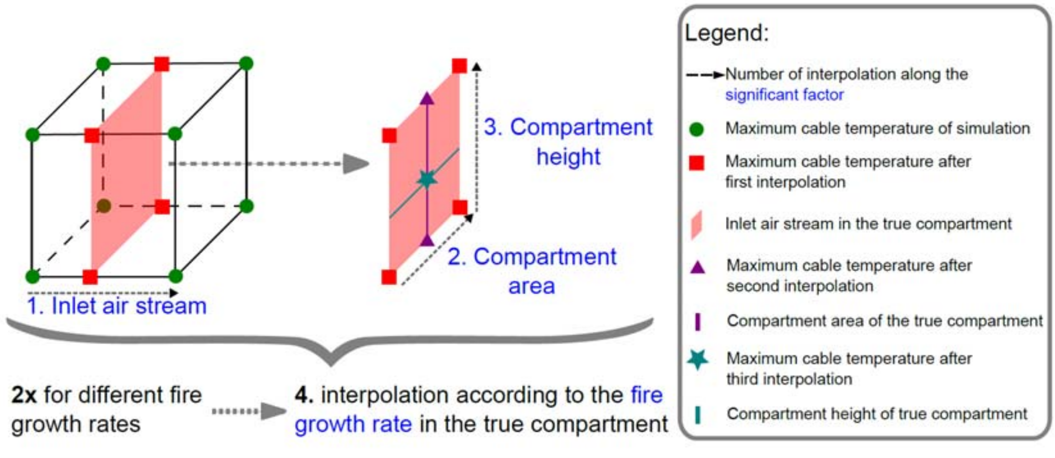


Figure 3. **Linear interpolation of the maximum cable temperature along the four dimensions of the significant factors according to the compartment configuration of the true compartment to be screened**

The method interpolates the simulation results of the fictitious compartments according to the compartment configuration of a true compartment in a NPP that has to be screened in the Fire PSA. Therefore, the method chooses eight simulations which have smaller and higher values in the three dimensions of the following significant factors compared to the true compartment: the inlet air stream, the compartment area, and the compartment height. The maximum cable temperature is interpolated, or if necessary extrapolated, according to the true compartment configuration along these three dimensions. This step is done twice for different fire growth rates. Finally, the maximum cable temperature is interpolated in the fourth dimension from these two results according to the fire growth rate of the true compartment. Hence, the result is the maximum cable temperature gained from 16 simulations that have significant factors close to the true compartment to be screened.

2.3 Definition of uncertainties and probabilistic modelling

The exact maximum cable temperature after the interpolation represents rather a distribution of possible maximum cable temperatures which can be expected in the true compartment. The distribution is caused by uncertainties in the fire model and the fire scenario. Additionally, experimental uncertainties lead to a distribution of the failure temperature of the reference cable. Thus, it is not possible to determine the occurrence of an electrical cable failure precisely but rather by a failure probability.

This subsection describes the determination of the failure probability. For this purpose, the random variables of the factor δ, of the failure temperature and of the fire growth rate are defined in the next three paragraphs and finally summarised in Table 2. Afterwards, two methods to calculate the failure probability are outlined, namely the First Order Reliability Method (FORM) and the Conditional Expectation Method (CEM). Finally, the last paragraph in this subsection describes how the set of filter criteria can be defined based on the failure probability of a true compartment to be screened.

The factor $\delta \approx -0.2$ and the Equation 1 express the underestimation of the maximum cable temperature in the simulations compared to the experimental results. The causes of the uncertainty of factor δ are uncertain conditions in the experiments used in the validation study. The largest uncertainty in the validation study lies in the measurement of the HRR. The HRR was determined by two different methods in the experiments, namely the oxygen consumption calorimetry and by the mass loss rate of the fuel, which lead to two different experimental HRR curves. Both HRR curves were modelled in two simulations for the one experiment and the effect on the maximum cable temperature was analysed. Hence, two δ's can be determined for two simulated maximum cable temperatures and one experimental cable temperature. Finally, derived from two experiments, the uncertainty of the factor δ caused by the uncertainty of the HRR is $\delta \approx -0.2 \pm 0.07$. There is a uniform distribution of the random variable δ between the minimum and maximum value. More information on this analysis are summarised in [15].

As mentioned in Subsection 2.1, the failure temperature T_{fail} of the reference cable was determined in three experiments [3, 4]. Since the failure temperature depends on several probabilistic factors like material properties, the structure of the cable as well the thermal decomposition of the insulation, the failure temperature was considered as lognormal distributed. The mean of the distribution was 240 °C. The 1% and the 99% quantiles were assumed to be 200 °C and 280 °C according to the experimental results, which leads to a standard deviation of about 17 °C.

The fire growth rate α was another source of uncertainty in the methodology. Since there were only few information on probabilistic distributions of the fire growth rate of cables in literature, information on the fire growth rate for this study were derived from experiments of the 'Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE)' [16]. Therefore, HRRs of 19 fires on horizontal cable trays with comparable layouts in eleven experiments were investigated. Different cables were installed on the cable trays as it is the case in NPPs. The fire growth rate α was analysed manually for each cable tray. The probability distribution of the fire growth rates fitted well to a lognormal distribution and the lognormal distribution fitted best to the experimental data with a mean of 0.00171 kW/s^2 and a standard deviation of 0.00192 kW/s^2.

Table 2 summarises the three random variables. The following two methods were used to include the random variables in the methodology and to calculate the failure probability: the First Order Reliability Method (FORM) and the Conditional Expectation Method (CEM) [17]. Both methods are outlined in the following two paragraphs.

Table 2. **Random variables used in the probabilistic set of filter criteria with their minimum (Min) and maximum (Max) values respectively their mean and standard deviation (StDev)**

Random variable	Distribution	Min / Mean	Max / StDev
Model uncertainty δ	Uniform	-0.27	-0.13
Failure temperature T_{fail}	Lognormal	240 °C	16.8 °C
Fire growth rate α	Lognormal	0.00171 kW/s²	0.00192 kW/s²

The FORM uses the first degree of the Taylor series of the two random variables fire growth rate and failure temperature [17]. Higher degrees of the Taylor series are not applicable here because of the linear interpolation of the simulation results. The FORM treats all random variables as normal distributions. For this reason, it does not consider the uncertainty of the factor δ. Consequently, the FORM causes a systematic error in the calculations of the failure probability but it is able to estimate quickly the approximate failure probabilities. Thus, FORM allows a quick overview over a large number of compartment configurations.

The CEM allows the more precise analysis of the failure probabilities compared to FORM. The CEM is a kind of Monte Carlo Simulation with one random variable as 'control variable' [17]. Here, the control variable is the failure temperature. The CEM calculates the maximum cable temperature in the compartment configuration in each simulation step and determines the corresponding failure probability using the probability distribution of the control variable. The final failure probability p_{fail} can be averaged from all simulation steps at the end of the Monte Carlo Simulation. The CEM requires far less steps to get accurate results compared to common Monte Carlo Simulations and has moreover no restrictions concerning the application of random variables as it is the case with FORM. Hence, the error in the results of CEM is small compared to the FORM but CEM requires still more time to calculate the failure probability. Accordingly, CEM is only applied on particular interesting compartment configurations to verify the results of FORM.

Finally, a true compartment can be screened out in Fire PSA if the failure probability of its specific compartment configuration exceeds a predefined accepted threshold value for the failure probability. In comparison to the 'fire load criterion', the methodology for the probabilistic set of filter criteria considers four significant factors to describe true compartments in NPPs. Furthermore, it has no restrictions on the local distribution of the fire load due to conservative assumptions in the compartment arrangements. However, the methodology covers only the cold gas layer and parts of the hot gas layer in a compartment fire (see Figure 4).

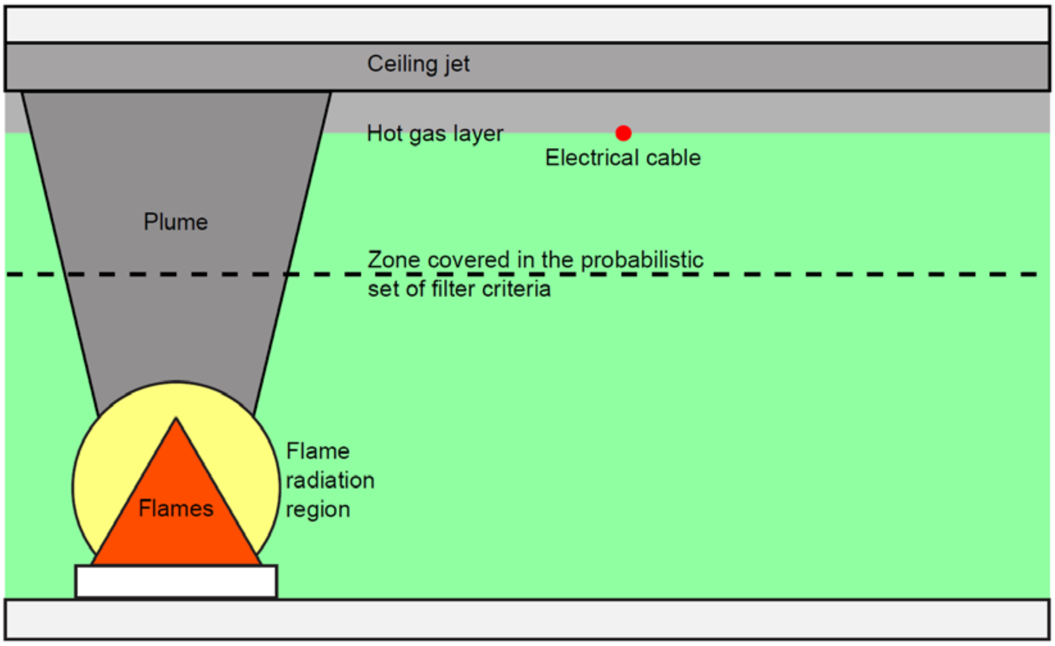


Figure 4. **The zone covered in the probabilistic set of filter criteria (green) together with the zones of influence according to [18]**

Components located in the hot gas layer closer to the ceiling as the electrical cable as well as components located in other 'Zones of Influence' [18] with a higher thermal impact, like the flame region or the ceiling jet, have to be analysed separately. Additionally, the set of filter criteria is not valid for non-thermal damages in particular of electronic components.

3. Effects of the Compartment Configuration on the Fire Behaviour and the Cable Temperature

The parameter study provides information on the effects of the significant factors on the fire behaviour as well as on the air and cable temperatures. In brief, all fire simulations with mechanical ventilation revealed qualitative comparable fire behaviour. To illustrate this, the air and cable temperatures of two simulations are exemplary plotted in Figure 5. The compartment in both simulations had a size of 196 m², a height of 5 m and a fire growth rate of 0.005210 kW/s² which is between slow and moderate. The only difference between the two simulations was the inlet air stream which was 0.5 m³/s (Sim. 1) as well as 2 m³/s (Sim. 2).

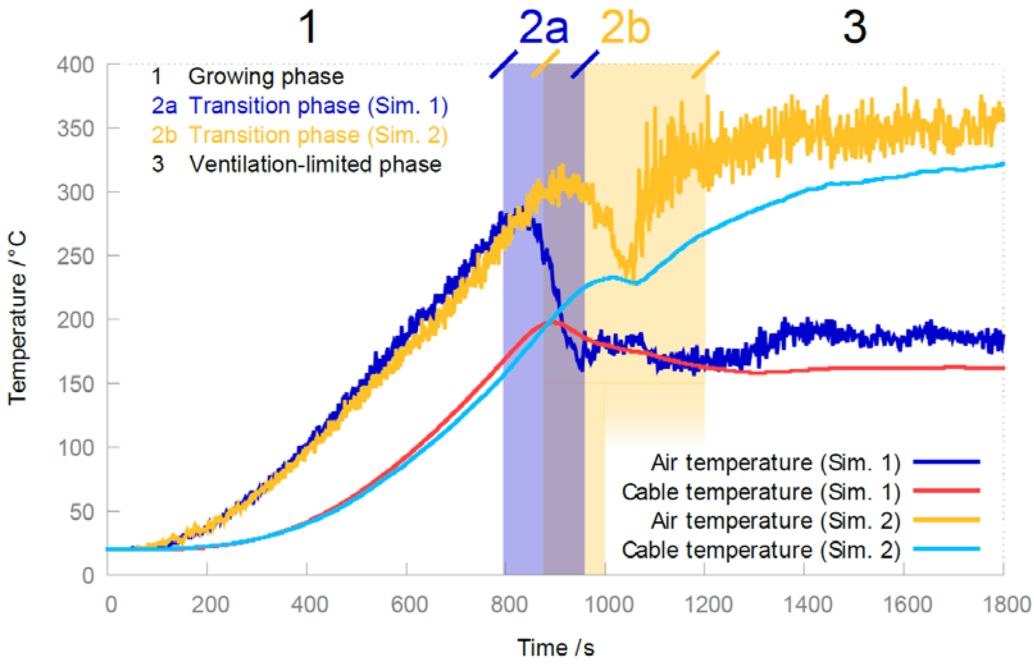


Figure 5. **Air and cable temperatures during two simulations with different inlet air streams**

Generally, the fire behaviour splits up in three consecutive phases: 1. the growing phase, 2. the transition phase (2a and 2b in the Figure 5 for both simulations) and 3. the ventilation-limited phase. The air temperatures in the growing phase of the fire followed the t-squared approach of the prescribed HRR. The HRR reaches the transition phase as soon as the oxygen in the compartment is consumed and the HRR merges into the ventilation-limited phase. The HRR and the air temperatures during the ventilation-limited phase are nearly constant. As expected, the cable temperatures follow the air temperatures during the whole simulations. Since the HRR in the ventilation-limited phase correlates with the volume flow of the inlet air stream, simulation 2 leads to higher air and cable temperatures compared to simulation 1.

The fire changes its behaviour during the transition phase in all simulations with mechanical ventilation. More precisely, the fire extinguishes on top of the fire source but fuel is still released due to the prescribed HRR. Hence, the fuel spreads throughout the compartment and finally burns close to the inlet air vents because of steady oxygen supply. The underlying sub-model in FDS for this behaviour is the extinction model by Mowrer [12]. Naturally under-ventilated small and real-scale fires showed comparable behaviour in several tests [19, 20] but the corresponding behaviour was not reported under mechanically ventilated conditions. Thus, the sub-model of Mowrer seems not to reflect entirely the real physical processes but the fire behaviour observed in the simulations is conservative.

Fires in compartment configurations with no mechanical ventilation show a different behaviour. Since the outlet air vents were realised as open vents for pressure balance in the compartment, FDS simulates alternately an inflow and an outflow depending on the pressure in the compartment. The inflow entrains fresh air in the compartment and leads to a spontaneous combustion of unburned fuel at the outlet air vents until the pressure increase generates an outflow. This leads to combustion with a strongly oscillating HRR at the outlet air vents after the transition phase. The fire would presumably extinguish in reality because of different boundary conditions at the outlet air vents. Based on this assumption this phenomenon is conservative too.

In general, the inlet air stream shows the strongest impact on the maximum cable temperatures. The maximum cable temperatures of all simulations are plotted against the maximum HRR for different inlet air streams in Figure 6. High inlet air streams (4 m³/s) lead in general to high maximum HRRs and high maximum cable temperatures (Region 1a in Figure 6). Some exceptions were at 1.4 MW and 3.2 MW (Regions 1b and 1c in Figure 6) where the fire growth rate was too slow to develop under-ventilated conditions until the end of the simulations. Other significant factors play only a minor role at high inlet air streams.

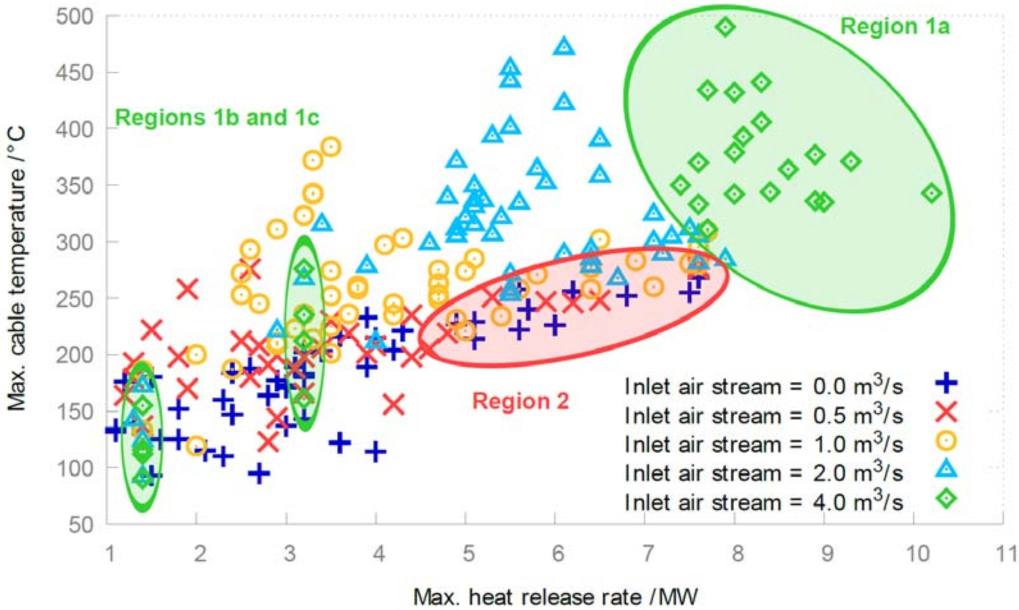


Figure 6. **Correlation between maximum (max.) HRR and maximum cable temperature for different inlet air streams**

Compartment configurations with no or low inlet air streams (0.5 m³/s) can also lead to high HRRs because of oxygen supply by large compartment volumes (Regions 2 in Figure 6). But the high HRRs do not lead to high maximum cable temperatures in these cases because the heat is dissipated in the entire compartment volume. In summary, the inlet air stream is the most important significant factor followed by the compartment size and height. The fire growth rate has only minor influence on the maximum cable temperature but can be important in small compartments with a low inlet air stream.

4. Theoretical Application of the Probabilistic Set of Filter Criteria on a Compartment

A compartment described in a study of the project BMU 2005-665 [11] was chosen as theoretical example. The compartment to be theoretically screened had an area of approximately 138 m² and a height of 4.6 m. The ventilation was either turned off or turned on with an air change rate of 5 h^{-1} which corresponds to an inlet air stream of about 0.884 m³/s. In contrast to the BMU project [11], in this example the fire growth rate α was used from fires on horizontal cable trays based on the experimental results as shown in Subsection 2.3 that was thus much slower than the value of 0.0833 kW/s² used in [11]). The maximum HRR was determined by the two-step reaction and extinction model of FDS and differs from the value of 3 MW in the BMU project [11]. The reference cable was calculated in 5.6 m horizontal distance to the fire source and 1 m below the ceiling. Figure 7 shows the effect of the random variables on the maximum cable temperature for the compartment with an air change rate of 1.5 h^{-1}.

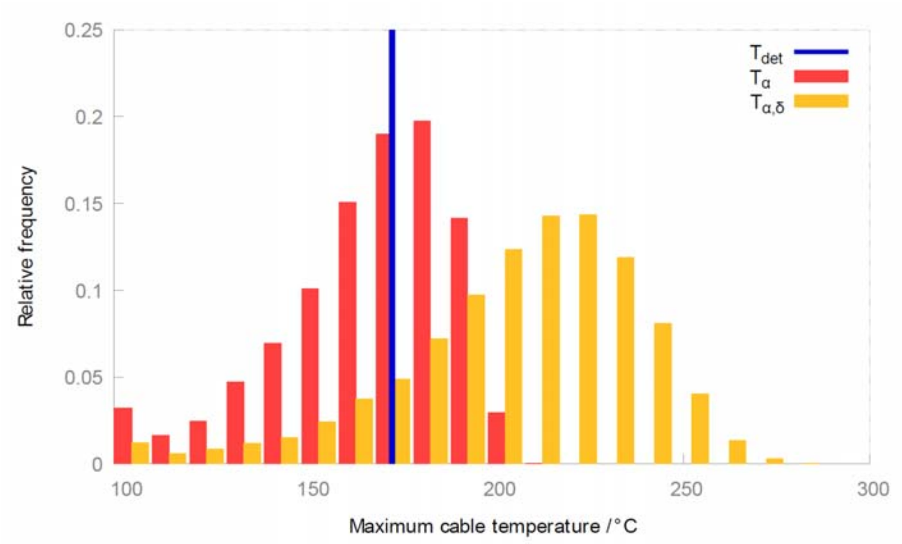


Figure 7. **Differences between the deterministic maximum cable temperature T_{det}, the distribution T^*_{α} considering the effect of the stochastic fire growth rate, and the distribution $T^*_{\alpha,\delta}$ considering additionally the factor δ**

The deterministic result of the maximum cable temperature after the 4D-linear interpolation according to the compartment configuration of the example was T_{det} = 172 °C. This calculation considers the mean of the fire growth rate as a discrete value. Furthermore, the calculation does not take into account the underestimation of FDS of the maximum cable temperatures by the factor δ. The red (left) distribution of the maximum cable temperature T^*_{α} shows the effect of the lognormal distribution of the fire growth rate α. The yellow (right) distribution of the maximum cable temperature $T^*_{\alpha,\delta}$ additionally considers the effect of the factor δ and thus represents the maximum cable temperature that could be conservatively expected in reality in a compartment fire with the same compartment configuration. The mean of $T^*_{\alpha,\delta}$ is 40 °C higher compared to the mean of T^*_{α} due to the underestimation of the maximum cable temperature of FDS. The standard deviation of $T^*_{\alpha,\delta}$ was about 30% higher compared to T^*_{α} because of the uncertainty in the factor δ. In conclusion, a deterministic set of filter criteria that does not consider the random variables would screen out more compartments compared to the probabilistic set of filter criteria.

The air change rate varied between 0 h^{-1} and 5 h^{-1} in the example of the BMU project [11]. The FORM was used to calculate the probability of failure for different ventilation conditions. See Table 3 for the results for the compartment configuration of the theoretical example with different air change rates.

Table 3. **Comparison between the failure probabilities calculated by CEM and FORM for the compartment configuration of the theoretical example with different air change rates**

Air Change Rate /h^{-1}	$p_{fail,FORM}$	$p_{fail,CEM}$
0	0.00	0.01
1	0.06	0.05
1.2	0.10	0.07
1.5	0.17	0.10
2	0.35	0.18
5	1.00	0.89

Assuming an accepted failure probability of e.g. $p_{fail,acc} = 0.1$, an air change rate of 1.2 h^{-1} seems to be critical to screen out the compartment of the theoretical example according to the results of FORM. The failure probability was verified by the CEM with 5000 Monte Carlo iterations. The result of the CEM is $p_{fail} = 0.068$. The air change rate of 1.5 h^{-1} leads to a failure probability of $p_{fail} \approx 0.1$ according to CEM. In conclusion, the FORM is a good approximation for low failure probabilities and thus can be used to get an overview about a lot of compartment configurations in a NPP.

The probabilistic set of filter criteria could be extended by an additional filter criterion, which uses the fire load criterion as basis. A total heat of approximately 700 MJ was released by the fire until the electrical cable failure occurred in the simulations of the given example. If the compartment contained less fuel, the fire would have extinguished before the electrical cable failure occurred. Hence, the critical fire load density of this example is approximately 5 MJ/m². Other compartment configurations are in the same order of magnitude. This result contradicts the validity of the fire load criterion since it is much lower compared to the 90 MJ/m². However, a critical fire load density of 5 MJ/m² seems to be too low to serve as additional filter criterion.

5. Conclusion and Outlook

A probabilistic set of filter criteria was developed to screen out compartments from precise analysis in Fire PSA. The probabilistic set of filter criteria comprises four significant factors, which are crucial for the damage of components caused by fires in a compartment. Thus, the probabilistic set of filter criteria provides a comprehensible justification for the screening of compartments and is more specific to compartments compared to the fire load criterion.

Some improvements can be made before the practical application in Fire PSA: First, more simulations in the parameter study would decrease the systematic uncertainty caused by the linear interpolation of the simulation results. A four dimensional response surface, represented by a scalar field, could summarise all simulation results and facilitate the data access on the simulation results compared to the four-dimensional linear interpolation. Second, the simulations in the parameter study could consider more than one reference cable on more locations compared to the simulations presented in this paper. And third, additional criteria for other zones of influence could be implemented. These improvements would extent the applicability of the probabilistic set of filter criteria.

Additionally, further research should be performed in order to increase the accuracy of the results. The simulation of under-ventilated fires could be improved for large-scale applications. Some conservative fire phenomena, like the combustion at the inlet and outlet air vents, could presumably be removed. Further validation studies on the fire model in under-ventilated conditions is another important issue to reduce the uncertainty in the factor δ. Furthermore, additional experiments on the failure temperature of the reference cable could decrease the deviation of the failure probability and therewith increase the accuracy of the results.

References

[1] Türschmann, M., von Linden, J., Röwekamp, M. (2005), Systematisches Auswahlverfahren für probabilistische Brandanalysen, BMU-2005-667, Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, Berlin (Germany), 2005

[2] Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke (2005), Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-37/05, Bundesamt für Strahlenschutz (BfS), Salzgitter (Germany), 2005

[3] Iqbal ,N., Salley, M. H. (2004), Fire Dynamics Tools: Quantitative Fire Hazard Analysis Methods for the U.S. Nucleary Commission Fire Inspection Program, NUREG-1805, U.S. Nuclear Regulatory Commission, Rockville (Maryland USA), 2004

[4] Hosser, D., Riese, O., Klingenberg, M. (2005), Durchführung von Weiterführenden Kabelbrandversuchen einschließlich der Präsentation der Ergebnisse im Rahmen des Internationalen Projektes ICFMP, BMU-2005-663, Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, Berlin (Germany), 2005

[5] McGrattan, K. (2007), Cable Response to Live Fire (CAROLFIRE), Volume 3: Thermally-Induced Electrical Failure (THIEF) Model, NUREG/CR-6931, Vol. 3, U.S. Nuclear Regulatory Commission, Rockville (Maryland USA), 2007

[6] Melis, S., Audouin, L. (2008), Effects of Vitiation on the Heat Release Rate in Mechanically Ventilated Compartment Fires, Fire Safety Science-Proceedings of the ninth international Symposium 9:931–942

[7] Hosser, D. (2009), Leitfaden Ingenieurmethoden des Brandschutzes, Vereinigung zur Förderung des Deutschen Brandschutzes e.V., Altenberge (Germany), 2009

[8] McGrattan, K., McDermot, R., Hostikka, S., Floyd, J. (2010), Fire Dynamics Simulator (Version 5), User's Guide, National Institute of Standards and Technology (NIST), Gaithersburg (USA), 2010

[9] Lee, Y.-H., Kim, J. H., Yang, J. E. (2010), Application of the CFAST zone model to the Fire PSA, Nuclear Engineering and Design April:1–6

[10] Salley, M. H., Kassawara, R. P. (2010), Nuclear Power Plant Fire Modeling Application Guide (Draft Report for Comment), NUREG-1934, U.S. Nuclear Regulatory Commission, Rockville (Maryland USA), 2010

[11] Röwekamp, M., Heitsch, M., Klein-Hessling, W. (2005), Erste Benchmark-Rechnungen im Rahmen des International Collaborative Project to Evaluate Fire Models for NPP Applications, BMU-2005-665, Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), Berlin (Germany), 2005

[12] McGrattan, K., Baum, H., Rehm, R., Mell ,W., et al. (2010), Fire Dynamics Simulator (Version 5), Technical Reference Guide, Volume 1: Mathematical Model, National Institute of Standards and Technology (NIST), Gaithersburg (USA), 2010

[13] Audouin, L., Chandra, L., Consalvi, J.-L., Gay, L., et al. (2011), Quantifying differences between computational results and measurements in the case of a large-scale well-confined fire scenario, Nuclear Engineering and Design 241:18–31, 2011

[14] Hamins, A., Maranghides, A., Johnsson, R., Donnelly, M., et al. (2005), Report of Experimental Results for the International Fire Model Benchmarking and Validation Exercise 3, NIST Special Publication 1013-1, National Institute of Standards and Technology (NIST), Gaithersburg (USA), 2005

[15] Berchtold, F., Forell, B., Krause, U. (2011), Fire Dynamic Criteria for Compartment Screening in the Frame of Fire PSA, in: Proceedings of SMiRT 21, 12th International Pre-Conference Seminar on "Fire Safety in Nuclear Power Plants and Installations", München, Germany, September 13-15, 2011, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, GRS-A-3651, September 2011 177–194, 2011

[16] McGrattan, K., Lock, A., Marsh, N., Nyden, N., et al. (2010), Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Volume 1: Horizontal Trays, Draft Report for Comment, U.S. Nuclear Regulatory Commission, Rockville (Maryland USA), 2010

[17] Ayyub, B., McCuen, R. (2003), Probability, Statistics, and Reliability for Engineers and Scientists, Second Edition, CRC Press LLC, Boca Raton (USA), 2003

[18] Kassawara, R. P., Hyslop, J. S. (2005), EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850, U.S. Nuclear Regulatory Commission, Rockville (Maryland USA), 2005

[19] Gottuk, D. T., Lattimer, B. Y. (2003), 'Effect of Combustion Conditions on Species Production', In: Chapter 2-5, SFPE Handbook of Fire Protection Engineering, Third Edition, Society of Fire Protection Engineers, Bethesda (Maryland USA), 2003

[20] Utiskul, U., Quintiere, J., Rangwala, A., Ringwelski, B., et al. (2005), Compartment fire phenomena under limited ventilation, Fire Safety Journal 2:367-390, 2005

Fire PSA Automation Tool

Albert Malkhasyan
Tractebel Engineering S.A., Belgium

Abstract

The significant risk contribution from internal fire events observed at various NPPs across the world necessitated a systematic and detailed evaluation of fire events within PSAs to understand their importance and significance. This led to evolution of some state-of-the-art guidelines and standards (NUREG/CR-6850 and ASME PRA Standard) for performance of Fire Probabilistic Safety Assessment whose focus has also been to avoid the unrealistic conservatism present in existing methods that potentially result in overestimation of the fire risk at NPPs.

While these standards provide well structured and comprehensive methodological steps to fire PSA, they however demand handling of huge amount of data and significant efforts to perform the analysis required within each methodological step and for iterations between the various steps. For handling of the huge input data, complex databases are used and for handling the complex interlinks of cross-dependent tasks, potential use of numerous modelling and calculation software for fire modelling, risk quantification, Bayesian inference, etc., becomes essential. These aspects make it practically difficult to manage and optimize the time needed to perform Fire PSA based on the latest guidelines.

Tractebel Engineering has foreseen these difficulties while planning to perform the detailed Fire PSA for the Belgian NPPs as part of a regulatory requirement and proposed to develop a tool to make this task more efficient and optimised in terms of both time and resources.

This paper describes the automatic algorithm that is developed by Tractebel Engineering S.A. in connection with their fire PSA tool “VULCAN” for obtaining reasonably realistic results from Fire PSA study in a reasonable timeframe. The algorithm includes data processing from the input databases up to the final quantification of fire PSA event trees. The tool is connected with RiskSpectrum using the latest excel import and export feature in RiskSpectrum. The automatic algorithm is also designed to perform simple fire modelling for the chosen number of compartments with an automatic generation of fire scenarios as an input to the RiskSpectrum PSA model.

1. Background

The Probabilistic Safety Assessment (PSA) as well as the operating experience shows the importance of fire events on overall plant safety. The state-of-the-art methodology of Fire PSA (NUREG/CR-6850) demands huge amount of time and resources as the methodology involves handling of large amounts of input data and different types of analysis to be carried out. This is foreseen to result in delays in obtaining Fire PSA related plant improvement recommendations, and huge efforts in keeping the desirable level of the analysis detail.

To cope with these efforts, an initiative has been taken by Tractebel Engineering S.A. (Tractebel) to develop a computerized tool that can facilitate a Fire PSA study based on the state-of-the-art methodology. The tool is aimed to provide significant reduction in the amount of time required and improve the efficiency in handling the analysis requirements within each step and interlinks between various methodological steps.

The general technical aim of the tool is to automatize the qualitative and quantitative parts related to the Fire PSA quantification. Implemented in the tool at present is a simple fire propagation modelling, which can be easily replaced by more accurate results using more detailed fire modelling software, when available. TE is proposing to use this tool for performance of Fire PSA for the Belgian NPPs (which is now a regulatory requirement in Belgium).

2. Overview of Fire PSA Tool "VULCAN"

The Fire PSA computerized tool developed by Tractebel called "VULCAN" is an Access database driven tool that contains data tables with input and output data, as well as intermediate results for data traceability and reporting purposes. The data tables have complex database relationships between each other. The Access database performs data processing using SQL and VBA programming languages, and is capable of obtaining up to the final quantification results of the Fire PSA study.

A key feature of the tool is based on the new functionality of RiskSpectrum PSA, which allows Import and Export of Excel tables and reports for reading/modifying existing RiskSpectrum PSA model. This option allows automatic modification of the RiskSpectrum model by extracting, modifying and uploading back the model using Microsoft Excel. This functionality is used to perform several stages (iterations) of Fire PSA model quantification.

The final result of the analysis with VULCAN represents a CDF (core damage frequency) from fire initiating events based on the input data collected and processed by the tool working in conjunction with RiskSpectrum. VULCAN incorporates all tasks that are included in a Fire PSA study based on the NUREG/CR-6850 methodology [1].

Some tasks are modelled through a single VULCAN input data sheet that represents summarized results of that task. The other tasks are modelled partially or fully in VULCAN itself using programming languages. The level of modelling in VULCAN depended on the possibility to automatize certain steps given the initial data.

Functionally VULCAN consists of the core (Access database driven tool) that contains subsequent macros manipulating the data using SQL and VBA programming languages. Whenever needed VULCAN core calls and runs Excel VBA macros and other Access sessions, which perform particular analysis and data processing. In addition VULCAN core also interacts with RiskSpectrum software by providing input and obtaining quantification results.

3. Input Data Requirements

VULCAN works with predefined input data format that incorporates data from plant walkdowns and cable routing.

Plant walkdown information includes detailed information on fire loads, their locations and properties. It also includes general information on plant compartments, boundaries, fire extinction systems and equipment located inside those boundaries. Missing or incorrect data is treated by making assumptions during the manual data preparation phase and/or data processing phase performed within VULCAN. The advantage of manual data preparation and check is that the potential errors in the input data are acknowledged and treated correspondingly. This will also help during the verification of the results.

Availability of the input data in predefined format is a key requirement for the Fire PSA analysis using VULCAN. In most of the cases missing information will lead to over-conservatism of the results For instance missing data on object location within a compartment may lead to permanent failure of particular equipment given a fire inside that compartment. This will be the case even if in reality the equipment will not be affected by the fire scenario. Therefore, in order to benefit from the tool, it is essential to collect proper input data in a predefined format. This has been achieved by using another database developed for tracking walkdown and cable routing data.

Following are the input data required for the Fire PSA analysis using VULCAN:

- Information on ignition sources, targets (PSA equipment and cables) and other fire loads including their thermo-physical properties and their exact 3D location
- Information on compartments/rooms including their dimensions, surface and volume
- Detailed information on compartment/room boundaries such as:
  - Floors/ceilings
  - Walls
  - Doors
  - Penetrations
  - Other holes
- Information on ventilation of each compartment/room
- Information on fire detection and extinction means
- Additional detailed information on ignition sources and fire loads:
  - Cable trays, cable shafts, wiring
  - Electrical engines (pumps, valves, compressor motors, etc.)
  - Electrical cabinets, boxes, other electrical equipment
  - Other liquid, solid and gas fire loads
- Object dimensions and 3D localization is an essential information

A special attention should be given to the data related to the cable routing. Exact information about location of the cable trays within compartment is an essential data that ensures the quality of the analysis. Particularly any cable segment or cable tray should be traced as accurately as possible. This is to say, they should be divided into sub-segments (horizontal and vertical) describing that cable segment or cable tray as good as possible, with regard to its physical location and orientation within the room.

4. Data Quality Needs

The data quality will have a significant impact on the functioning of the VULCAN and its results. Depending on the data quality and precision (related to each input information required) different effects on the results are expected. Some of those effects are discussed below.

4.1 Cable trays

The major effect on the results is expected to be from the data collected on cable trays. Depending on how exact the information on the location of cable trays is, the fire modelling results could be too conservative.

If a single cable tray has a complicate tracing path and has not been subdivided into sub-trays with location details of each of them, it could get modelled in a very conservative way.

If a cable tray is represented with one coordinate position (e.g. the gravity centre of the cable tray) and the cable tray orientation has not been collected, the cable tray will be modelled as a sphere with the diameter equal to the cable tray length. In case of long cable trays, this will result in very conservative results, assuming that cable tray can be potentially located almost everywhere in the given room.

4.2 Fire loads/targets

Another issue is the orientation of large fire loads or targets. In most of the cases fire loads and targets are small objects of standard dimensions (e.g. electrical cabinet, box, electrical motor, etc.). Assumptions made regarding orientation of those objects will have no noticeable influence on fire modelling calculations. However there could be several objects of large irregular sizes (i.e. Length >> Width), where the absence of object orientation could potentially have a conservative impact on the results with regard to that object.

4.3 Virtual wall

In order to specify more precisely location of components in large compartments, these compartments were subdivided into smaller rooms (with a unique plant labelling) that were bounded by so-called “Virtual wall”. “Virtual wall” represents a virtual object that connects two locations (rooms) together. “Virtual wall” also is used for cases when two rooms were connected through large openings allowing unimpeded fire propagation between those rooms. In all cases “virtual wall” represents nothing but large openings, and therefore has no fire resistance.

Considering software limitation together with restrictions related to its runtime, simplifications were introduced concerning fire propagation that was limited to a few number of rooms delimited by “Virtual wall”. For big compartments, such as Turbine Hall or Reactor Building, presence of “virtual wall” could erroneously limit fire propagation within those compartments if those compartments are not be modelled properly in the PSA.

4.4 Matching of a fire load with PSA basic event names

In order to perform the Fire PSA it is necessary to identify all important Fire PSA safety equipment and obtain their location and details. Thus it is essential that the matching between fire loads found during walkdown and the Fire PSA safety equipment is ensured. Otherwise failure of equipment during fire scenarios will not be reflected in the PSA model and as a consequence the PSA results will be optimistic.

4.5 Equipment type correspondence

Each plant equipment analysed in VULCAN should be associated with each of the following elements:

- fire ignition frequency
- thermo-physical properties
- physical location and object dimensions

For each of the above mentioned elements a specific list of equipment type is foreseen. The specific list of ignition sources types (bins) is suggested by NUREG/CR-6850 methodology (§6.3.1) [1]. This list may combine different plant equipment types into a single bin. Similar approach was used for categorization of fire loads for walkdown, which provides the location and dimensions of each fire load.

A one-to-one correspondence between all equipment types should be ensured. Omission of any correspondence between those equipment types may result in optimistic effect in the analysis. For instance equipment may have zero frequency or zero maximal heat release rate during fire scenario modelling.

5. Methodological Steps in VULCAN

The NUREG/CR-6850 methodology [1] formed the basis for the development of the fire PSA tool VULCAN. All tasks with the exception of the seismic-fire interaction and the uncertainty analysis were incorporated in VULCAN.

Section 5.1 to 5.13 describe in detail how each task of the methodology has been incorporated in the tool. It should be noted that due to the graded nature of the Fire PSA methodology certain tasks do not necessarily require to be completed. For instance, cable circuit analysis is not necessarily needed for obtaining a final CDF value. Conservatively cable impact may be considered for all selected cables. Similarly human reliability analysis can be performed only with one level of detail (e.g. quantitative screening). Therefore scoping and detailed human reliability analysis may be omitted during data processing and taken up later on. This allows flexibility in step by step development of a complete Fire PSA analysis and the results can be quickly recalculated after any change in the import datasheets.

5.1 Plant boundary definition and partitioning

Plant boundary definition and partitioning is performed by PSA experts with help of fire hazard analysis results. The results of the plant boundary partitioning are incorporated into an import data table that contains information on the building, its compartments and rooms included in those compartments. The approach followed in VULCAN does not consider fire propagation from one compartment into the other. Therefore the plant boundary partitioning should justify each compartment.

5.2 Fire PSA component selection

The component selection and the selection of potential initiating events due to fire is performed analytically by PSA experts. The results are prepared in a specific format and are uploaded into VULCAN using manual tables.

Potential initiating events are associated either with a set of equipment, or with certain rooms. The latter is used in case there is a procedure that requires plant shutdown following a fire in a particular room. But it can be also used to simplify the definition of initiating event scenario. For instance an initiating event “loss of certain system” can be modelled by assigning initiating event to a set of rooms instead of a set of equipment.

5.3 Fire PRA cable selection

Cable selection is performed analytically by PSA experts. The results are prepared in a specific format and are uploaded into VULCAN using manual tables. The results include both cables needed for Fire PSA equipment functioning and control as well as cables needed for operator actions.

Cable circuit failure analysis can be performed before or after this step. However at this step the failure of a particular cable will be associated with all the basic events linked to the equipment that involves a given cable.

5.4 Qualitative screening

This task is automatically performed by VULCAN by selecting only those compartments that contain Fire PSA safety equipment or their associated cable. Therefore information on equipment and cable location is essential to perform this task.

5.5 Fire-induced risk model

This task is automatically performed by VULCAN. The tool automatically choses the appropriate event tree to represent the fire scenario based on the damaged equipment set. If there is a choice between few possible event trees to be analysed, the tool will analyse all cases separately or will choose the most conservative one, depending on the user's choice. The analysis of initiating event combination is not foreseen by VULCAN. Therefore it is up to the analyst to identify any new combinations of initiating events and to include them manually in the PSA model before running VULCAN.

5.6 Fire ignition frequencies

This task is done partially by the PSA expert and then automatically finalized by VULCAN using manual tables. The fire frequency analysis is a plant specific task. Depending on data availability different types of data sources maybe used. Therefore different type of data analysis will be applied on the proposed data sources. For the Belgian plants a common method was applied prior to importing the data into VULCAN.

Three types of manual data tables are required by VULCAN to finalize this task. The first table includes the ignition source bins with the plant-specific frequencies that were obtained following the data analysis of the fire events. Similarly to the NUREG/CR-6850 [1] the ignition source bin frequency represents a frequency per ignition source type (set).

The second table includes the list of all the transient fires in combination with all the plant rooms. Each combination should have a prepared list of all weighting factors as mentioned in the NUREG/CR-6850 methodology [1].

The third table is an essential part of VULCAN providing the link between the ignition source bins, the fire loads observed during the walkdowns and the plant equipment. As already discussed in §4.5 any missing link in this table will influence the results.

The final fire frequencies per ignition source are automatically calculated by VULCAN based on the ignition source count within a single ignition source bin.

5.7 Quantitative screening

This task is done automatically by VULCAN. Two iterations of quantitative screening are foreseen in VULCAN. Each iteration includes the automatic creation of new fire scenarios, the assignment of parameter values and the quantification of those fire scenarios. This is done using Excel I/O Risk Spectrum feature. In both iterations the input data is prepared on a compartment basis. The Fire PSA model built at an earlier step (see §5.5), is substituted with fire scenario frequencies and human error probabilities.

The first iteration, performed after the task Fire ignition frequencies includes all basic events related to failed equipment and cables that are turned to "True". In the second iteration, performed after the task Scoping fire modelling, the fire ignition frequencies are adjusted according to the results of the calculations of the severity factors (following Scoping fire modelling; see §5.8) and the list of basic events is adjusted accordingly. VULCAN performs all these adjustments automatically.

As the result of this task the fire scenarios are given together with their core damage frequencies. This is the basis for selecting the top significant fire compartments (scenarios). The criterion used for the selection of the most significant scenarios is flexible in VULCAN. Depending on the final CDF target, the objectives of the study and the available resources for performing the refining calculations the criterion can vary within a wide range of values. As suggested by the NUREG/CR-6850 methodology [1] the total contribution to the CDF was selected as a criterion. In practice VULCAN can also run without such a criterion. That is to say, all the compartments can be considered in each step of the analysis up to the end of the analysis. This is up to PSA expert to choose the appropriate criteria.

5.8 Scoping fire modelling

This task is done automatically by VULCAN. The fire modelling represents one of the most complex tasks in the VULCAN. The credibility of the results is highly dependent on the input data precision and quality. Deterministic calculation provides the results in the “digital” format, e.g. “equipment failed” or “equipment not affected”, therefore depending on the kind of imprecision results may have both optimistic and conservative results. In case of doubt the results should be checked and corresponding changes to the input data have to be made.

The scoping fire modelling is performed based on the ZOI (zone of influence) approach discussed in the NUREG/CR-6850 methodology [1]. The compartment is split between the following zones of influence: flame, plume, ceiling jet, smoke and flame irradiation zone. The latter is conservatively assumed to be everywhere in the compartment (e.g. the equipment is always analysed for the radiation heat transfer from the flame).

If an equipment is located at the border of a particular ZOI it is considered to be in that ZOI. Conservatively the target temperature is taken equal to the corresponding ZOI temperature. Therefore simple temperature equations, similar to the ones used in FDT [3, 4] and FIVE [5] methodologies, are used to determine if the equipment will be considered damaged given the maximal heat release rate of the ignition source. Both temperature and radiant heating criteria is used to determine if the equipment is damaged.

The first aim of this task in VULCAN is to study each and every ignition source for having a possibility to affect any target or fire load within the same compartment. If this is not the case (e.g. ignition source does not have any impact on any fire load or target) the ignition source will be screened out (This corresponds to have a severity factor set to zero). The second aim is to estimate the severity factor for all ignition sources that were not screened out during this task.

The fire severity factor is then taken into account in the frequency estimation of each ignition source and therefore in the compartment fire frequency.

5.9 Detailed circuit failure analysis

As this task requires a detailed analysis of electrical circuits, it will be done by I&C experts with assistance from PSA experts. The results of this task are optional and are imported to VULCAN using manual tables.

VULCAN provides an extract with the list of cables to be analysed in detail. This list depends on the compartments selected following the first quantitative screening (see §5.7). The results of the analysis, a list of the Fire PSA safety equipment basic events retained due to cable failure and their associated failure mode, are given back to VULCAN using manual tables. If there are no results to be imported by VULCAN, by default VULCAN will consider the entire list of associated basic event postulated for all the cable failure modes.

5.10 Circuit failure mode likelihood analysis

This task is also foreseen to be executed by I&C and PSA experts similarly to the previous task (see §5.9). The approach for providing extracts and data import is the same as for the detailed circuit failure analysis. The results of this task are also optional.

The aim of this task is to obtain likelihood probabilities for each cable failure mode. Therefore this is the only additional information required by VULCAN. By default the probability considered in the study will be P=1, which is equivalent to the fact that the task has not been performed.

5.11 Detailed fire modelling

Detailed fire modelling is another very complex and time consuming part of VULCAN that is executed fully automatic. The core of calculations is based on the equations used in the Scoping fire modelling task (see §5.8). Those equations are used to determine the fire scenario – a time dependent sequence of fire loads and targets catching fire and failing. In this context a fire load represents any item that can catch fire and start generating heat, whereas the target is an equipment (not necessarily a fire load) that will have an impact on the fire scenario modelled in the PSA (i.e. a PSA equipment). A non-suppression probability is used to determine the frequency of failing certain target set at different time steps.

The damage criterion for a target is the same as the ignition criterion for a fire load. This assumption is a quite significant assumption that may lead to large conservatism in the study. In many cases the damage criteria will be smaller than the ignition criteria of the same equipment (e.g. pump motor damage temperature and pump motor ignition temperature).

The modelling of the fire scenario is done in the following way:

- ignition source affects the most vulnerable fire load
- the effect on all adjacent fire loads (targets) is considered from both the ignition source and the fire load at the same time
- the ignition source together with the fire load affect another most vulnerable fire load
- the effect on all adjacent fire loads (targets) is considered from all fire loads (involved up to this time step in the fire) at the same time

The fire scenario stops in the following cases:

- there are no more fire loads/targets left to catch fire (damage)
- there are still several fire loads/targets left, but the fire has been neutralized by itself (i.e. the amount of flammable material has been burnt out)
- it is possible to consider a low frequency threshold to stop the fire scenario simulation
  - the best criteria would be the consideration of low CDF impact of the scenario, however this option will slow down the assessment significantly (even comparing to the complete scenario runtime)

Some assumptions have been taken also for the fire propagation from one room to another within a compartment. The fire propagation is performed by heat transfer from a particular room to adjacent ones through walls and openings, and by flame irradiation through doors and holes.

As already mentioned the fire propagation between fire compartments is not taken into account (see §5.1), as long as it has been justified in fire hazard analysis that the fire has no capability to propagate further.

The output results of this task are the target sets with corresponding frequencies that take into account the non-extinguishing probability. The target sets for a single fire scenario are defined in the following manner:

- ignition source (time = 0)
- ignition source + fire load 1 (time = T1)
- ignition source + fire load 1 + fire load 2 (time = T2)
- ignition source + fire load 1 + fire load 2 + target 1 (time = T3)
- ignition source + fire load 1 + fire load 2 + target 1 + … (time = T4)
- etc (time = TX)

Considering that the non-extinguishing probability at each time-step (T1, T2, …) are equal to P1, P2, P3, P4, …, PX, the frequency of each target set (F1, F2, …) will be calculated the following way:

- FX = F (ignition source) * PX
- …
- F3 = F (ignition source) * P3 – SUM (F4, …, FX)
- F2 = F (ignition source) * P2 – SUM (F3, …, FX)
- F1 = F (ignition source) * P1 – SUM (F2, …, FX)
- F0 = F (ignition source) * 1 – SUM (F1, …, FX)

Note that there is the following correlation between the non-extinguishing probabilities for a given fire scenario: PX < … < P4 < P3 < P2 < P1 < 1. Subtraction of frequencies of bigger target sets (e.g. SUM (F3, …, FX)) is conditioned by the fact that:

- the sum of all frequencies in a single fire scenario should be equal to the frequency of the particular ignition source (F0 + F1 + … + FX = F (ignition source))
- all smaller target sets are fully included in the bigger target sets, making a part of them

The target sets are grouped together for the same “compartment – initiating event” combination by summing up the total frequency. The grouping is performed rather based on the basic events sets rather than the equipment sets. This is done in order to avoid duplicated scenarios in the Fire PSA model. Summing up similar sequences helps to keep those scenarios above the cut-off limit. Not grouping of similar scenarios may potentially mask some moderately important sequences.

5.12 Post-fire human reliability analysis

This task is performed partially by VULCAN and needs input from a PSA expert by using several manual tables. Particularly screening and scoping human reliability analyses (HRA) for existing human actions are adapted to be fully incorporated in VULCAN. According to NUREG/CR-6850 [1] and NUREG-1921 [2] methodologies, it is required to assess the feasibility of the existing human actions and to categorize them into four different sets.

There are several qualitative criteria used for assigning some screening HRA probability sets. Those qualitative criteria have been transformed into tool algorithm. For instance the presence of spurious indication during certain human action is determined by observing the cable providing information to the operator in the list of damaged cables during a particular scenario. Another example is that significant environmental impact on the main control room crew is considered only to be within the compartment involving the main control room. For local actions reasonably conservative values have been established to take into account the delay in reaching the particular room.

For the purpose of the automatic performance of screening and scoping HRA a number of manual tables have been created. Those tables include information on the human action, the related emergency operating procedure and the associated steps, the indications and equipment associated with those operating procedures, the time available to perform the action in the internal event PSA model, etc.

Similarly to the screening HRA, an algorithm was created to perform the scoping HRA based on the existing manual tables. For the detailed HRA a specific manual table is foreseen to incorporate the updated human error probability values into the VULCAN.

5.13 Fire risk quantification

This task is the last automatized task in VULCAN. The aim of this task is to quantify the final frequencies of all the significant fire scenarios. The final CDF obtained as the result of VULCAN simulation is obtained by summing up the latest frequencies of all fire scenario core damages. This is to say, all the fire scenarios that were screened out during first quantitative screening will not be analysed in the second quantitative screening and the final quantification, but will be incorporated in the final CDF value.

6. Structure of VULCAN

Generally the VULCAN core can be represented as consecutive sub-codes (sub-macros) that implement different Fire PSA methodological steps. Following sub-macros are introduced:

- First macro, including Tasks 1, 2, 3 and 4 (from NUREG/CR-6850 [1]),
  - Uploads all required input data into Access database
  - Fills manual tables related to the fire boundary partitioning, component and cable selection
  - Performs qualitative analysis
  - For each fire compartment develops possible RiskSpectrum model options (different initiating events are possible)
  - Runs RiskSpectrum to identify the most conservative initiating event
- Second macro, including Tasks 5, 6, 12 and 7 (from NUREG/CR-6850 [1]),
  - Finalize the choice of the RiskSpectrum model for each fire compartment
  - Fills manual tables related to fire ignition frequencies
  - Performs human reliability screening analysis and obtains screening values for the human failure events
  - Prepares Fire PSA RiskSpectrum model for the first quantitative screening

  - Runs RiskSpectrum quantification
- Third macro, including Tasks 7, 8 and 9 (from NUREG/CR-6850 [1]),
  - Obtains results from RiskSpectrum
  - Selects the significant fire compartments for more detailed analysis
  - Generates scenarios for scoping fire analysis
  - For each selected compartment runs the fire scoping analysis separately for each ignition source
  - Generates the output table for cable circuit analysis
- Forth macro, including Tasks 6, 7, 8, 9 and 12 (from NUREG/CR-6850 [1]),
  - Obtains results from scoping fire analysis
  - Obtains results from cable circuit analysis (optional)
  - Recalculates the fire frequency for each compartment
  - Obtains new scoping values for the selected human failure events
  - Prepares the Fire PSA RiskSpectrum model for the second quantification
  - Runs RiskSpectrum quantification
- Fifth macro, including Tasks 7, 10, and 11 (from NUREG/CR-6850 [1]),
  - Obtains results from RiskSpectrum
  - Selects the significant fire compartments for more detailed analysis
  - Generates scenarios for the detailed fire analysis
  - For each selected compartment runs the detailed fire analysis separately for each ignition source
  - Generates (optional) output table for the cable circuit analysis (cable failure mode likelihood analysis)
- Sixth macro, including Tasks 6, 7, 10, 11, 12 and 14 (from NUREG/CR-6850 [1]),
  - Obtains results on detailed fire analysis
    - Either from the detailed fire analysis made with VULCAN algorithm
    - Or from the Excel files, resulting from detailed fire analysis made with 2D or 3D fire modelling software, in a predefined format.
  - Regroups all the target sets within the same “fire compartment - initiating event” combination
  - Obtains results from the cable circuit analysis (fills in corresponding manual table)
  - Recalculates the fire frequency for each fire scenario
  - Obtains new detailed values for the selected human failure events
  - Prepares the Fire PSA RiskSpectrum model for the final quantification
  - Runs RiskSpectrum quantification
  - Obtains the final CDF

These six sub-macros are executed subsequently and trigger other executables to perform fire modelling analysis and/or RiskSpectrum quantification.

All required intermediate results are stored in Access database and are available after VULCAN finishes all calculations.

7. Conclusion

This paper presented the attempt by TE to automatize some of the tasks in Fire Probabilistic Safety Assessment to effectively deal with the complexity and comprehensiveness of current state-of-the-art methodology. The idea of automation evolved at TE following a regulatory requirement in Belgium to implement Fire PSA for all Belgian NPPs based on the state-of-the-art methodology. The idea picked up momentum after the new functionality of RiskSpectrum PSA to import and export excel reports has been released.

The main steps that were automated are related to qualitative and quantitative analysis, but also to the simplified fire modelling that will allow obtaining realistic results while staying conservative in basic assumptions. The other tasks were partially automated together with the import of some preliminary results into VULCAN in a predefined format.

The Fire PSA automation tool VULCAN requires input information in specific formats. Good quality of that information is essential to profit from the tool in a realistic way. VULCAN potentially includes certain types of assumptions that can be refined by customising the tool and by increasing the precision of the input data.

References

[1] EPRI/NRC-RES, NUREG/CR-6850, Fire PRA Methodology for Nuclear Power Facilities Volume 2: Detailed Methodology, EPRI 1011989, September 2005

[2] EPRI/NRC-RES, NUREG-1921, Fire Human Reliability Analysis Guidelines, EPRI 1023001, July 2012

[3] U.S. NRC, NUREG-1805, Fire Dynamics Tools. Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, December 2004

[4] U.S. NRC, NUREG-1824, Volume 3, Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications: Fire Dynamics Tools (FDT), May 2007

[5] U.S. NRC, NUREG-1824, Volume 4, Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications: Fire-Induced Vulnerability Evaluation (FIVE), May 2007

Simulation Toolbox Development for Fire PRA

Simo Hostikka*, Johan Mangs, Terhi Kling, Antti Paajanen,
Timo Korhonen, Tatu Pinomaa
VTT Technical Research Centre of Finland, Finland
*) Current address: Aalto University, Espoo, Finland

Abstract

In this presentation we summarize the development work for the numerical simulation capabilities of fire development and effects. The developed topics were chosen to fulfil the observed gaps in our capability to evaluate the defence-in-depth principle in fire situations. The topics therefore include the models for the fire burning rate, thermally-induced failures of fire barriers and the subsequent failure of compartmentation, smoke-induced failures on electronics, and the performance assessment of the responding organizations.

1. Introduction

The numerical fire simulation techniques are nowadays used to support the quantitative fire-PRA in order to i) increase the understanding on the dynamics of the underlying fire phenomena, ii) to reduce the conservatism, and iii) to find out the most significant sources of risk. For many years, we have applied the deterministic fire simulation tools within a probabilistic framework to calculate the conditional probabilities of the fire consequences. These techniques have successfully been applied in the PRA work of the Finnish NPPs. The interesting consequence type has usually been a thermal damage of electrical or other devices. Less attention has been paid on the other types of failures, such as smoke-induced damage of electronics, and the temporal development of the fire hazards and its coupling with the response of the plant personnel and safety systems.

In this presentation, we summarize the simulation tools development work carried out in the Finnish nuclear power plant fire safety projects. The aim of the work is a validated capability to evaluate the fire-defence-in-depth. We used this aim to choose the topics of research. The simulation chain starts from the specification or prediction of the fire heat release rate. For this reason, the condensed-phase fuel models are developed for the FDS fire model. Next, the simulations describe the heat and mass transport phenomena inside the compartment of fire origin and the connected compartments. From the PRA viewpoint, however, it is often more interesting to study the probability of fire spreading to neighbouring compartments which can belong to another, redundant sub-system. Therefore, we have considered the assessment of the fire-barrier performance and structural response. We have already included the active suppression systems in the numerical and stochastic simulations for some time, but it has been more difficult to evaluate the operational effectiveness of the fire brigade and other participating organizations. We therefore developed a new model for the simulation of the operation time of the responding organization. Finally, the fire-induced failures to the technical systems need to be evaluated. For this purpose, we have started to build a simulation chain for smoke-induced failures.

2. Simulation of fire loads

2.1 Pool fires

Pools of combustible liquids are one of the most significant potential fire loads in nuclear facilities. The consequences of the liquid pool fires are commonly assessed by specifying the pool burning rate using empirical correlations or data, and by using either hand calculations or CFD to calculate the resulting thermal impacts. In either case, the burning rate of the liquid pool needs to be estimated. Our aim has been to develop a necessary sub-routine to FDS program to predict the liquid pool evaporation rate and its dependence on thermal and chemical conditions. We discuss this topic in a separate presentation and paper of the workshop (Sikanen).

2.2 Cable fires

Burning and thermal failures of electrical cables have been one of the most important research topics for many years, mainly because the cable fire loads dominate the fire risk of many commercial power plants. The goal of the research has been a validated capability to simulate the burning rate of cable bundles and trays under external or internal ignition scenario and with and without the presence of active fire suppression systems. The numerical modelling relies heavily on experiments in determining the input parameters for the thermal and chemical sub-models, as well as validation. Recently, the work has focussed on the determination of thermal decomposition kinetics and combustible product yields from thermo-gravimetric analysis (TGA) and Micro-scale Combustion Calorimetry (MCC). Means to model the individual cables as sub-grid scale elements of the CFD model have been studied, and the results are validated using U.S.NRC Christifire [1] data and the recent results from the OECD PRISME2 –project. More details can be found from a separate presentation and paper of the workshop (Matala & Hostikka).

3. Fire barrier performance assessment

3.1 Procedure for barrier failure probability

3.1.1 Modelling principle

Performance of safety barriers can be characterized by functionality/effectiveness, reliability/availability, response time and robustness or by the triggering event or condition in which the barrier is needed. The overall performance of the fire safety design is determined by qualitative or quantitative assessment with deterministic/probabilistic analyses. Qualitative assessment takes care that all the relevant fire safety measures are in use and well qualified to be able to prevent the fire propagation through the levels of defence in depth. Quantitative assessment gives us information of the fire consequences as well as the risks associated to it.

In NPPs the safety blocks (parallel systems) have to be separated to their own fire compartments as far as possible. The confinement of a fire inside an area is justified when the duration and thermal conditions of the hypothetical fires are less severe than what the fire barriers can resist, in other words their fire performance. This is typically based on conventional theoretical fire duration and thermal stress curves. According to NUREG CR-6850 [2] there is no specific method for modelling the barrier response in actual fire scenarios that is suitable for incorporation into a Fire PRA. In the case of passive fire barriers, the current methods rely on “qualification”, a process that prescribes design, installation and maintenance of the fire barrier in accordance with an approved fire protection program.

The new method that we suggest for further development and testing is based on the basic ideas of the EPRESSI-method [3] that is developed for the evaluation of a sufficient fire resistance performance of fire barriers in the context of new NPPs. In this new method, the compartmenting components are virtually tested by models that are calibrated according to the original fire test results that have been used to determine the rating of the components. From this virtual testing, the components get performance curves that are compared with Monte Carlo fire simulation results. The previous barriers of the defence in depth fire safety solution (Figure 1) can be included in the fire simulation, so that the final compartmentation step can be tested by the new method.

Fire-separating building elements and the associated equipment and fittings must be made so that the spread of fire from one department to another within a specified period of time will be prevented (E1/E2 building code of Finland). Fire rating of the parts of the building is the time in minutes that the part can fulfil the required criteria when exposed to the fire load according to the standard time-temperature curve (ISO 834). The standard curve can often be considered overly conservative as it is based on experiments with flashover. Another problem is that with the standard test considers only fires that last a limited time. Real fires can have different time-temperature curves.

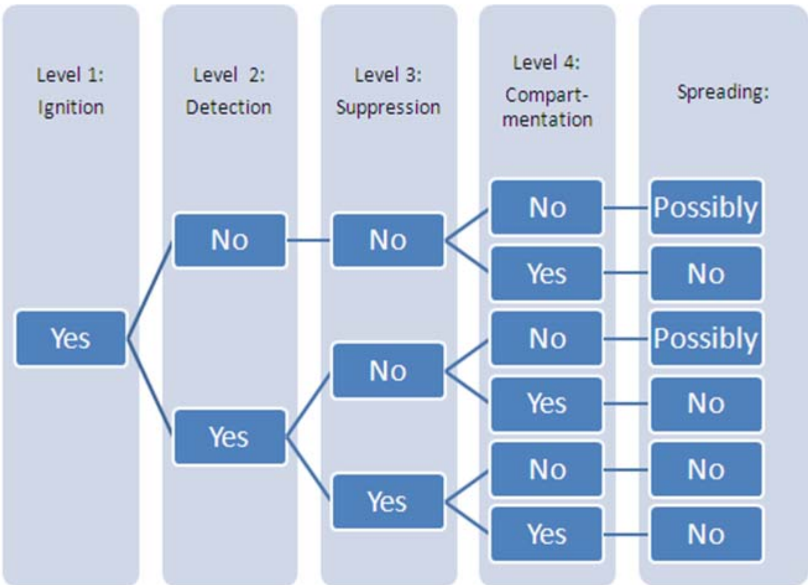


Figure 1. **Fire defence in depth – event tree for the possibility of fire spreading from a compartment to another.**

Based on the basic ideas of EPRESSI-method [3] we suggest a method that is based on three main steps:

1. Determine the performance curves of the compartmenting components.
2. Determine the fire load curves in the compartment.
3. Test how well the components resist the fire loads.

With the method we will be able to evaluate the compartmentation performance, i.e. the 4th level of the fire defence in depth in Figure 1. The possibility of fire spreading from one compartment to another does not automatically mean that a fire would spread. It means that this case should be further studied: Even if the compartmentation would fail all fires do not have a potential for spreading or making damage in another compartment.

We determine the performance curves of the compartmenting components in the following way: First, we simulate the fire resistance tests with applicable models. Examples of the modelling alternatives are listed in Table 1. The models are calibrated with test data. Next, we use the calibrated models to determine the resistance of the components for time-temperature curves (test loads) I=1…n that are usually different than the standard ISO 834 curve. We consider the components qualified for those time-temperature curves, for which the rejection criterion (Table 1) is not met. These curves are called 'performance curves'. At the moment, the rejection criteria are usually based on temperature because the determination of the structural response and the corresponding criteria would be an extremely complicated task. Coupled heat-transfer simulations are already being made but the extension of the analyses to mechanical response is still difficult. An example of the result is shown in Table 2 for a fictitious set of components and performance curves.

Table 1. **Modelling the fire resistance tests (examples).**

Component	**Typical rejection criterion (Gautier et al., 2010)**	**Modelling**
Door	$\Delta T > 140°C$ on the unexposed surface of the leaf	1D (FDS)
Damper	$\Delta T > 180°C$ on the anchoring	1D (FDS)
Penetration seal	$\Delta T > 180°C$ on the surface of the seal (side not exposed to the fire)	3D (FEA)
Wrapping	$\Delta T = 188°C$ inside the sheath	2D (FEA)
Housing	$\Delta T = 180°C$ on the unexposed surface of the housing ($\Delta t = 160°C$ inside the housing)	1D (FDS)
	$\Delta T > 188°C$ ambient in the housing 1350 mm above the floor	2D (FEA)
Ventilation duct	$\Delta T > 140°C$ on the section of ducting 25 mm from the bead	2D (FEA)

Table 2**. Virtual testing of the components (example).**

Test load (i)	i=1	i=2	i=3
Component 1	pass	Pass	pass
Component 2	fail	Fail	pass
Component 3	fail	Pass	pass

Next, we determine the temperature load curves of the room by Monte Carlo fire simulations. Suitable random variables in, for example a cable room, could be the size and location, the properties of the cables and the construction materials and the response of the sprinkler system. The reliability of fire detection and extinguishing systems as well as the operational fire-fighting effects could be connected to the fire simulation results to get the share of the fires that need to be used for the testing of the components.

For each component one needs to test if, for each simulated fire realization and for each performance curve i, the following holds:

$$T_{realization}(t) < T_{performance\ (i)}(t), \forall t \tag{1}$$

The share of the cases where clause (1) is true gives us the probability

$$P_{performance} = \frac{N_{true}}{N_{tot}}$$

where

$P_{performance}$ = probability that the compartmentation is realized for the individual component, when the component is in use ("the door is closed etc.").

N_{true} = number of the fire realizations, where clause (1) is true for at least one i

N_{tot} = total number of the fire realizations

The performance tree of the compartmentation is presented in Figure 2.

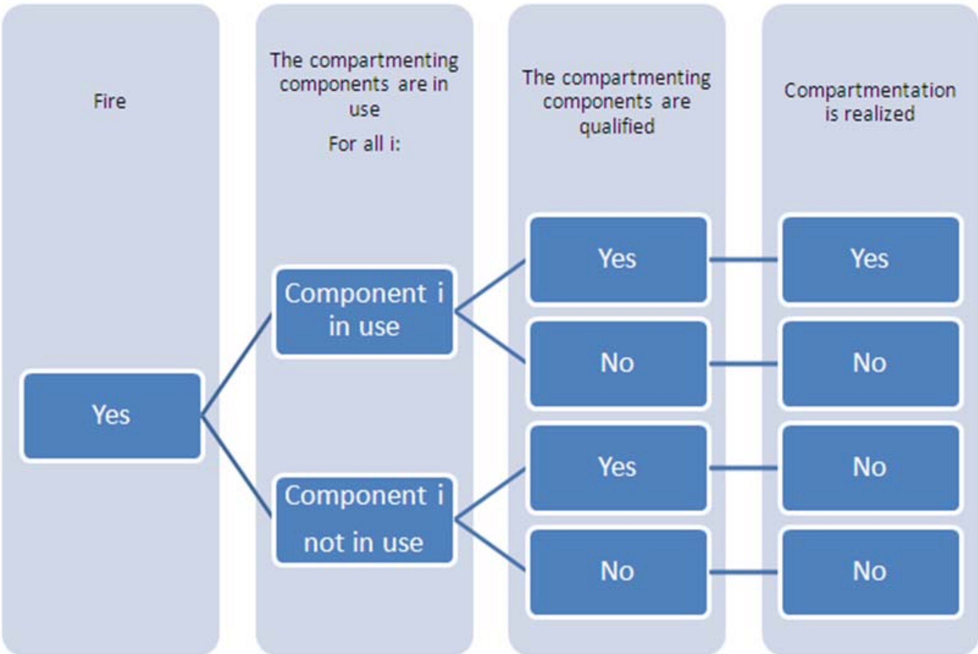


Figure 2. **Performance tree of the compartmentation.**

3.1.2 Modelling example

The fire resistance of an element is typically assessed by applying a temperature load corresponding to the standard ISO 834 time-temperature curve, and measuring the back side temperature of the element. However, real fire loads can significantly differ from the logarithmically increasing ISO 834 / EN 1363-1 curve which is defined as

$$T_{ISO}(t) = T_0 + 345 \log_{10}(8t+1) \quad (2)$$

where T_0 is the ambient temperature, 20 °C, and variable t is the time in minutes. As an example, the performance of a simple model door is considered under a family of piecewise linear fire loads that differ significantly from the ISO 834 curve. The studied performance criterion is heat insulation of the element, so that the door back side should not pass 160 °C.

We performed the simulations using a one-dimensional wall heat conduction model on the Fire Dynamics Simulator (FDS) code. We assumed that no chemical reactions or radiation take place inside the fire door. Thermal boundary condition on the door front side is the incident radiative heat flux of a black body radiator, with a predetermined time-temperature curve. The cooler back side uses convective cooling to room temperature of 20 °C. In these simulations, the door consists of 2 mm thick steel plates, separated by a 52 mm thick insulating (Rockwool) layer. The insulator thickness was adjusted to 52 mm, so that the door reaches 160 °C, when exposed to the ISO 834 fire load (Eq. 2).

Next, we exposed the model door to the fire curves sketched in Figure 3, for which the peak temperature T_p was set to 1100 °C, the maximum baseline temperature T_b to 500 °C, and ambient temperature T_a to 20 °C. The temperature peak width Δt_p was varied randomly and uniformly between 40 and 100 minutes, and correspondingly, peak occurrence time t_p between 10 and 240 minutes. It should be noted that the values and distributions of the parameters of these time-temperature curves were selected based on the stochastic fire simulations for a cable room [4] but they did not have a precise connection to real conditions. They merely provide a way to consider a larger number of non-standard fire loads.

Figure 4a shows the time-temperature curves of the door front surface, when exposed to the non-standard fire loads for 100 simulation cases, together with the ISO curve as a reference. The corresponding backside temperatures are drawn in Figure 4b, which shows that when the fire load's temperature peak occurs early, the back side temperature remains small. Also, if the temperature peak width was close to the minimum value considered, 40 minutes, the back side temperature did not reach the criterion temperature, 160 °C. Therefore, the main finding was that for the considered non-standard fire loads, only late occurring peaks with a sufficient peak width can lead to temperatures exceeding the performance criterion temperature.

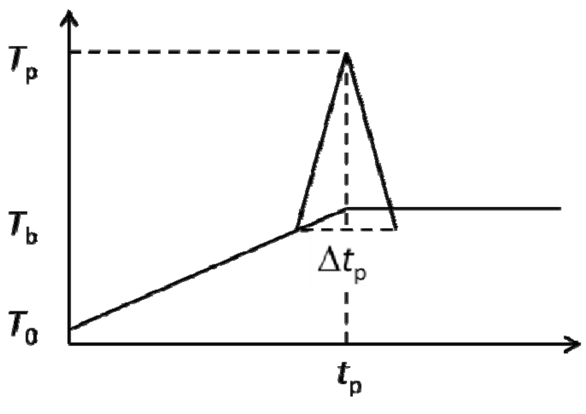


Figure 3. **Sketch of the studied non-standard time-temperature curve.**

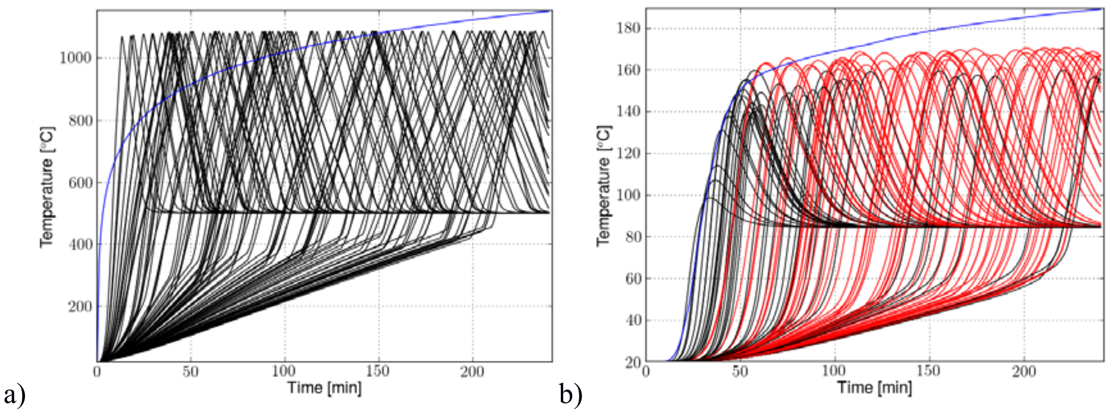


Figure 4: **Time-temperature curve in fictitious non-standard scenarios. a) Front temperatures for 100 and the ISO 834 curve (blue), and b) back side temperatures of the model door. Red curves are simulation cases for which the temperature exceeds 160 °C.**

3.2 Coupling between fire and structural simulations

Modelling the response of structural elements and the overall performance of larger structures under fire induced thermal and mechanical strains is a complicated modelling chain. The chain contains (1) modelling the fire environment, (2) modelling the thermal response of materials, (3) modelling the fire induced damage, and (4) modelling the response of the structures under loads. For this modelling task we have developed a FDS2FEM tool [5] that can be used to between the first and second steps in the chain. It provides a sequential one-directional coupling between a computational fluid dynamics (CFD) fire simulation programme FDS and some commonly used finite element (FEM) software suites (Currently supported are ABAQUS and ANSYS).

The coupling of the CFD fire simulation and FEM software could be done in many ways. The FDS2FEM coupling tool uses the simplest one where the fire model and the thermal model are decoupled by transferring appropriate boundary conditions from the fire simulation to the thermal model of the construction. This means that the surface temperatures and/or heat fluxes on the surfaces are transferred between the two models. This kind of decoupling is a commonly used among thermal models, i.e., there is no (thermal) feedback from the thermal response model to the fire environment. FDS2FEM is an interoperability tool between two different computational models that are working largely independently, just a minimal level of information exchange is needed.

The FDS2FEM coupling tool transfers thermal boundary conditions between the fire and thermal analysis programmes sequentially and one-directionally, i.e., the fire simulation is done first and its results are transferred to set of boundary conditions for the thermal analysis. Thus, we were able to implement the coupling tool as an external tool both FDS and FEM programmes. Different boundary conditions can be used, including the surface temperature, the net heat flux, and the adiabatic surface temperature. Adiabatic surface temperature (AST) offers a convenient way to transfer net heat flux from FDS to ABAQUS with minimal amount of information exchange [6]. It responds to radiative and convective heat fluxes with a negligible time lag. It is defined according to equation

$$\varepsilon\left(\dot{q}''_{r,in} - \sigma T_{AST}^4\right) + h\left(T_g - T_{AST}\right) = 0 \qquad (1)$$

where ε is the surface emissivity, $\dot{q}''_{r,in}$ is the incident radiative heat flux, h is the convective heat transfer coefficient and T_g is the local gas temperature. Thus, the net heat flux to a surface can be calculated within the FEM code as

$$\dot{q}''_{net} = \varepsilon\sigma\left(T^4_{AST,CFD} - T^4_{FEM}\right) + h\left(T_{AST,CFD} - T_{FEM}\right) = 0 \tag{2}$$

By using AST as the transfer quantity, we minimize the implicit effects of the FDS heat conduction solver.

The thermal boundary conditions obtained from the fire simulation are in the form of time series at FDS surface nodes that are faces of the rectangular obstructions in the FDS fluid calculation mesh. The FEM geometry usually differs from the FDS geometry due to the different modelling resolution. Often the FEM model has much finer details than the FDS model. Even if the geometries were identical in the models, the surface nodes would be different due to the different meshing schemes used in the programmes, as illustrated in Figure 5. A simple nearest neighbour mesh mapping method was adapted in the FDS2FEM tool with some additional features including, a cut-off radius, distance dependent weights and some robustness in the definition of the neighbourhood. The coupling tool is a command-line application for Linux and Windows operating systems.

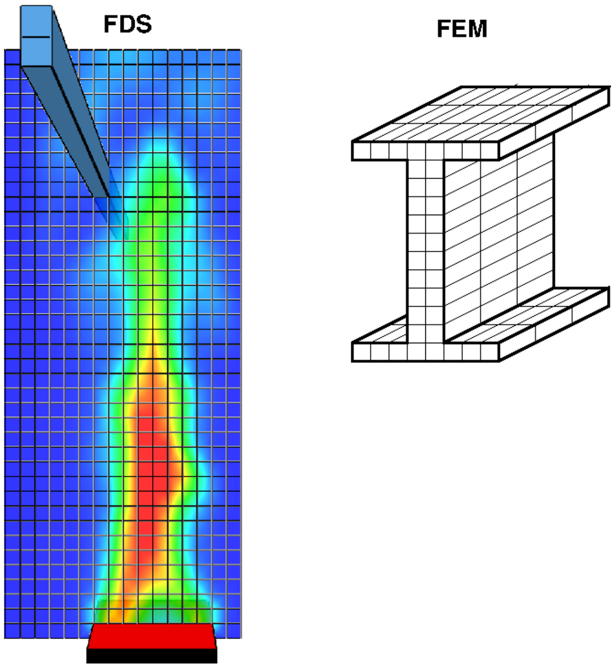


Figure 5. **A example of the FDS to FEM coupling case, where the different resolution of the models shows up**

4. Fire effects on electronics

Fires generate heat and smoke at different amounts. These products are possible causes of interruptions and malfunctions in electronics. As the nuclear power plants, both in Finland and worldwide, are replacing their old instrumentation and control systems with digital systems, and at the same time, new plants are being built with modern digital systems, the question arises: how does modern digital electronics respond when exposed to smoke and heat?

For thermal failures, the engineering assessment methods have been developed over the last few years. Most of these methods consist of the main steps such as fire source description or assessment, heat transport calculation, calculation of the target's thermal response and finally a comparison against empirically observed temperature criteria. We now propose an analogous methodology for soot-induced failures. The main steps are shown in Figure 6. They are:

1) Fire source and heat release rate assessment and specification.
2) Soot yield, and the consequent soot production assessment
3) Soot transport calculation, using e.g. CFD.
4) Soot deposition calculation, using e.g. CFD.
5) Assessment of the reduction in surface insulation resistance.
6) Comparison against critical threshold value for the insulation resistance.

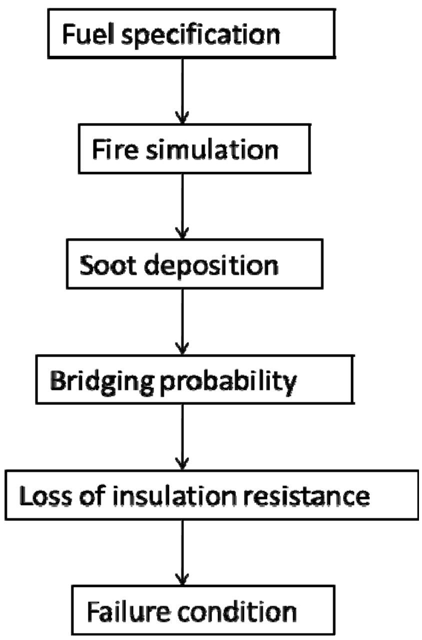


Figure 6. **Proposed methodology for soot-induced failures**

The purpose of our work has been to collect necessary input data and modelling tools for the complete assessment chain. We assumed that the heat and soot transport calculations will be carried out using the FDS code and its existing smoke deposition model [7]. The present version of FDS considers the following deposition mechanisms: gravitational settling, thermophoresis and combined diffusion-turbulence deposition. Electric fields can also contribute to deposition but are not considered in the present FDS. Dealing with soot effects on electronics, the electrophoretic deposition due to electric fields may also be important and was examined in this study. First, we collected soot yields for different fuels from small-scale tests such as the cone calorimeter, with emphasis on cable materials. Most experiments were for free burning at ambient conditions, and one should take care when using the values for smoke emission estimates because the size scale and the reduced ventilation can strongly affect the smoke production.

Next, we studied the main soot deposition mechanisms and soot properties affecting the deposition mechanisms and compared these mechanisms with electrophoresis to evaluate their relative importance in fire situations with electronics (Figure 7). We found that a relatively high electric field strength (100 kV/m or higher) was needed before the electrophoresis started to have significance on the soot deposition from a smoky environment. We therefore conclude that we can disregard the electrophoresis as a mechanism for soot deposition, and the present FDS soot deposition model is considered satisfactory.

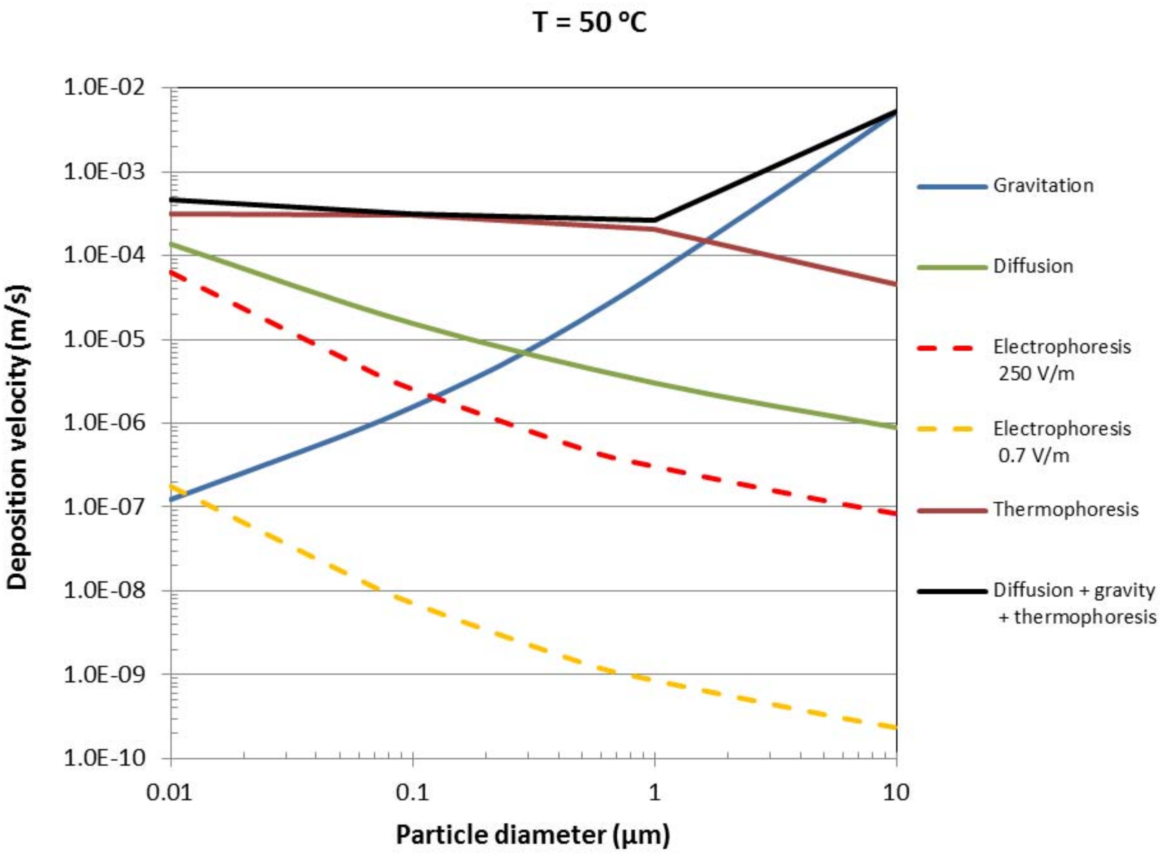


Figure 7. **Deposition velocities due to electrophoresis compared to other deposition mechanisms. Electric field strengths are literature values near electrical office equipment (250 V/m) or ambient in nuclear or electric power plants locations (0.7 V/m)**

Literature findings indicate that bias voltage on e.g. comb patterns can have influence on leakage currents by building strings of soot agglomerates between conductors, but the findings are indistinct. A short literature review on critical thresholds for circuit failure did not reveal any absolute correspondence between the change in surface insulation resistance due to smoke exposure and the failure of any electronic or electrical components. In special cases, approximate limits seem to be possible to estimate, and a 1 MΩ leakage resistance failure criteria has been reported in some studies [8]. Above this level less failures were noted, and for resistance below 1

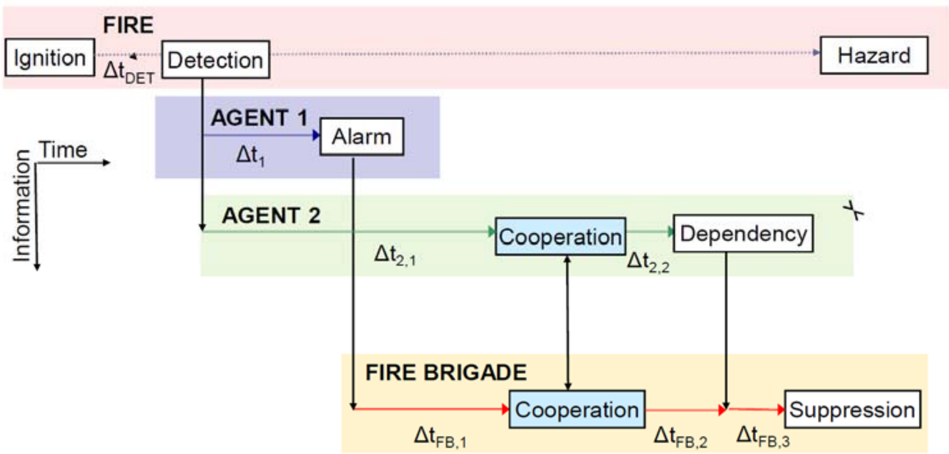
MΩ more failures were noted, but not in every test. Some models for estimating smoke damage potentials in industrial fire applications were found in literature [9], but their general validity should be further investigated. Overall, further efforts on steps 5 and 6 in the presented methodology sequence are needed.

5. Assessment of organization's response

As an attempt towards quantitative Fire-HRA, we have developed a computational model for the simulation of time delays concerning manual firefighting operations in a fire situation of a NPP. The model is called Stochastic Operation Time Model (SOTM) [4]. The main steps in creating a conventional fire-fighting related SOTM are listed below.

- Define a fire scenario and a corresponding fire-fighting scenario.
- Determine the actors and their connections.
- Analyse the action steps and possible deviations.
- Describe the operation time delays (in terms of probability distributions) for the action steps and additional delays (probabilities and probability distributions) in case of deviations.
- Carry out a stochastic analysis to get a probability distribution of the total time delay of the fire-fighting operations.

Before the quantitative analysis can be performed, the total time delay of the fire-brigade response has to be formulated as an analytical expression. Actions that take place sequentially (serial processes) are described by sums of time delays. When there are parallel actions needed to reach a common goal, the maximum delay is included in the sum. In a similar way, when there are alternative actions aiming at the same goal, the minimum time delay is included. The process of formulating the total delay is illustrated in Figure 8.



$$\Delta t_{OPER} = \Delta t_{DET} + \max\,[\Delta t_1 + \Delta t_{FB,1}\,,\ \Delta t_{2,1}] + \Delta t_{2,2} + \Delta t_{FB,3}$$

Figure 8. **A timeline showing the actors, connections and time delays in a generic fire brigade response scenario.**

In addition, there might be events and actions that only take place sometimes. We include these in the model as additional events that occur with a predefined probability. For example, the total time delay for serial processes with additional (serial) events can be written as

$$\Delta t_{tot} = \sum_i \left[\Delta t_i + \sum_j \left(k_{ij} \delta t_{ij} \right) \right] \tag{3}$$

where Δt_i is the time delay of event i, δt_{ij} is the time delay of additional event j related to event i, and k_{ij} is a random variable having value one or zero according to the probability p_{ij} of the additional event j.

Applications of the model have shown that formulating the total time delay of the fire-brigade response can be laborious and time consuming. We therefore started a further development of the SOTM methodology to speed up the modelling process, to increase the traceability and to reduce the vulnerability of the calculations. The model development presented here concentrates on defining general tools for easy implementation of new applications.

One alternative to formulating the total time delay as an analytical expression is to describe the actors along with their connections (i.e. interactions) and the associated time delays as an agent-based model, and let the actors interact in a continuous-time simulation. In this approach, the intellectual burden is shifted from formulating an analytical expression to creating a working and general-enough program logic and a feasible user interface. In a single application of SOTM, designing and programming a simulator is probably the more laborious option. However, once the basic features of the simulator are implemented, it can be used and re-used without much additional effort.

We used the NPP fire brigade response scenario described in [4] as a case study for creating a SOTM as an agent-based simulation. Figure 9 illustrates the fire-fighting scenario with the actors, connections and time delays.

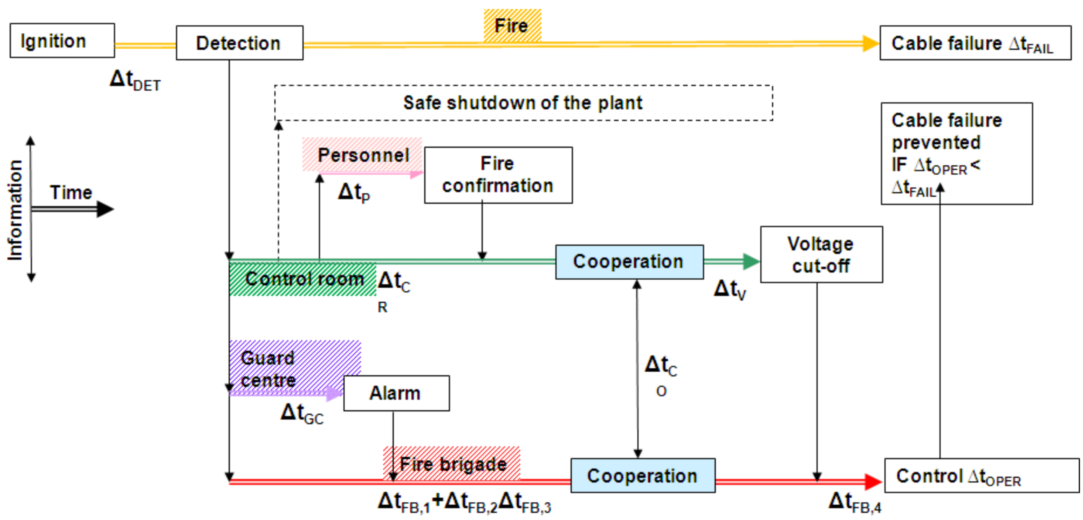


Figure 9. **Operational actions in a NPP cable room fire.**

We implemented the agent-based operation time model as a command-line operated C-program running a continuous-time simulation. The simulation set-up included four agents: the control room, the control room employee, the guard centre and the fire brigade. In addition, there were two system states: fire detection and cable failure. Each of the agents and system states — implemented in C as structs — were characterized by an internal state (integer) and one or more time delays (real numbers). For example, the guard centre was characterized by three possible internal states and a single time delay:

```
struct guard_centre {
  /* 0 = idle */
  /* 1 = alarmed */
  /* 2 = fire brigade alarmed */
  int state;
  /* Time required for alarming the fire brigade */
  double twait;
};
```

Rules for changing the internal state were formulated independently for each of the agents and the system states. Again, an example of the guard centre follows. Here, gc is a struct corresponding to the guard centre and t is the simulation time.

```
if (gc.state == 0) {
  gc.state=1;
  gc.twait=t+gc.twait;
} else if (gc.state == 1) {
  if (t >= gc.twait) {
    gc.state=2;
  }
}
```

We then created similar characterizations and state-change rules for the other agents and system states. A simulation run starts with fire detection changing its state from 0 to 1 and ends with fire brigade changing its state from 4 to 5 (i.e. from waiting for voltage cut-off to fire suppression). A command-line printout of an actual simulation is shown in Figure 10. The time delay distributions used in the simulation were arbitrary and only for demonstration purposes. This kind of simulation is extremely fast to execute, and can easily be repeated hundreds or thousands of times under a stochastic simulation framework.

```
$./sotm
--> Fire detected at time 134
Control room: receives alarm at time 134
Control room personnel: goes to confirm fire at time 134
Guard centre: receives alarm at time 134
Guard centre: alarms fire brigade at time 242
Fire brigade: receives alarm at time 242
Control room personnel: fire confirmed at time 517
Control room: fire confirmed at time 518
Fire brigade: arrives at location at time 704
Control room: co-operation with fire brigade begins at time 705
Fire brigade: co-operation with control room begins at time 705
Control room: voltage cut-off on preparation at time 831
Fire brigade: waiting for voltage cut off at time 831
Control room: voltage cut-off ready at time 1187
--> Fire brigade: fire suppression begins at time 1187
```

Figure 10. **Command-line printout of a trial simulation. Time delay distributions (as well as the time unit) chosen for the example are arbitrary.**

6. Summary and future work

For the last few years, we have developed the experimental and numerical simulation methods for the fire safety assessment of nuclear power plants. In this presentation, we summarized some of the recent developments that aim at forming a complete chain of simulation tools. The chain starts from modelling of fire development (i.e. HRR) on electrical cable and liquid pool fire loads. Other fire loads can, of course be specified as usual. Basic heat and mass transfer processes within the compartments are currently simulated using the fire CFD, and despite the specific needs of model development, these models were considered an available and validated capability. We have therefore focussed on developing new methods for the fire barrier performance assessment because the next step in the simulation chain is the spreading of fire from the compartment of ignition to neighbouring spaces. We have formulated a procedure for the barrier failure probability calculation by combining the stochastic fire simulations and the barrier response analyses. For the analyses, we have developed a new interoperability tool between the CFD fire simulation and finite element analyses of structures (FDS2FEM). We also sketched a modelling chain for the smoke-induced failures of electronics and started an effort to collect the necessary input data. Finally, we continued the development of the stochastic operation time model for the human and organizational response. Main development effort has been the usability improvement by implementing the model as an agent model instead of a flowchart model. Currently, an application case study is going on with a Finnish utility.

The future work in developing these tools should include:

- Further development and validation of the sub-grid scale modelling of cable objects and the associated heat transfer processes.
- Further validation of the liquid pool burning rate predictions in vitiated environments.
- Extension of the FDS2FEM tool for additional FEM tools and collection of user experiences for modelling the fire effects in real NPP structures.

- Models for the barrier components should be created and the validity of the temperature-rise failure criteria should be investigated.
- Prediction of the soot-induced surface resistance reduction on electronics and experimental determination of the statistically significant damage criteria should be carried out.
- The operation time model should include some basic models for the human decision making processes for the simulations of situations with limited instructions and procedures. We must also connect the agent-based operation time model to the stochastic modelling environment, i.e. the Probabilistic Fire Simulator (PFS).

References

[1] McGrattan, K., Lock, A., Marsh, N., Nyden, M., Bareham, S., Price, M., Morgan, A.B., Galaska, M., Schenck, K. Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 1: Horizontal Trays (NUREG/CR-7010, Volume 1). 2012.

[2] EPRI and U.S. NRC, 2005. EPRI/NRC-RES Fire PSA methodology for Nuclear Power facilities. EPRI 1011989, NUREG CR-6850.

[3] Gautier, B., Mosse, M. and Eynard, O., 2010. EPRESSI Method – Justification of the Fire Partitioning Elements. Sixth International Seminar on Fire and Explosion Hazards, Weetwood Hall, Leeds, UK, April 11th to 16th, 2010. Compartment Fires 2.

[4] Hostikka, S., Kling, T. & Paajanen, A. (2012), Simulation of fire behaviour and human operations using a new stochastic operation time model, 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012 (PSAM11 ESREL 2012)

[5] Paajanen, A., Korhonen, T., Sippola, M., Hostikka, S. Malendowski M., and Gutkin, R. FDS2FEM — a tool for coupling fire and structural analyses, in Proceedings of the IABSE Workshop Helsinki 2013: Safety, Failures and Robustness of Large Structures, Helsinki, 2013

[6] Wickström, U. Adiabatic surface temperature and the plate thermometer for calculating heat transfer and controlling fire resistance furnaces. Fire Safety Science 9 (2008), 330-338.

[7] Floyd, J.E., Overholt, K.J., and Ezekoye, O.A. Soot Deposition and Gravitational Settling Modeling and Impact of Particle Size and Agglomeration. Fire Safety Science – Proceedings of the Eleventh International Symposium, Christchurch, New Zealand, February 2014.

[8] Peacock, R.D., Cleary, T.G., Reneke, P.A. & Murphy, D.C. 2012. A literature review of the effects of fire smoke on electrical equipment. Engineering Laboratory, National Institute of Standards and Technology, 68 p. + App. 15 p. (NUREG/CR-7123).

[9] Newman, J.S., Su, P., Yee, G.G. & Chivukula, S. 2013a. Development of smoke corrosion and leakage current damage functions. Fire Safety Journal Vol. 61, p. 92–99.

Building up a Cable Management System and Its Use for Probabilistic Fire Risk Assessment

Andreas Hempel
KAB-tec GmbH, Company for Cable Management and Technical Consultancy,
Frankfurt, Germany

Abstract

Cable installations of nuclear power plants have constantly become larger and more complex during their lifetime. Already during the basic design phase, the designers recognized that the only reasonable way to economically and clearly arrange the required volume of cable connections was by means of computer aided planning. Accordingly, plant vendors developed different programs for the design of cable systems of nuclear power plants. But often, after the construction and commissioning phases of the plants, these programs and cable databases were not improved and maintained for their entire lifetime. Consequently, not all requirements of plant operators for project work, installation and safety analysis can be complied with. Furthermore, the fast development of database software and operating systems limits the period in which a legacy system can be used. To replace the legacy system, standardized software is the optimal solution because the prospective upgrades will be taken care of by the provider of the new system.

Nowadays, additional factors have to be taken into account in the planning and operation of cable systems of the plants. A couple of years ago, it was only necessary to provide information regarding the connection between feeders and components. Now, besides the augmented requirements for documentation, the need for further information regarding availability and security has come to the fore.

Constructors and operators of nuclear power plants need a continuous cable management system with graphical support for design, construction, operation and dismantling.

The cable management systems of different plants are used also for fire specific probabilistic risk assessment (Fire PRA). In several PRA projects it was identified that fire induced cable failures may constitute the dominant contribution to fire induced core damage frequency (CDF).

In the cable management system, all cables within the plant are recorded with identification number of cable, destinations, cable type, cable routing, and room numbers of the compartments passed through. Completed with the allocation data for cable number to component, a cable management system provides all necessary cable data for Fire PRA.

Analysis of the cabling of a nuclear power plant can provide important information for the sufficient modernization of cable systems, e.g. use of advanced cable types, separation and protection issues. Detailed information about all (especially safety relevant) cables in the NPP are vital for Fire PRA.

1. Introduction

Cable installations of nuclear power plants (NPPs) have constantly become larger and more complex during their lifetime. Already during the basic design phase, the designers recognized that the only reasonable way to economically and clearly arrange the volume of cable connections required was by means of computer aided planning. Table 1 shows the increase of the amount of cables from the end of the design phase of the plants to present (2013), caused by modifications, upgrading, uprating and retrofits. These examples are meant to demonstrate how the modification service records and documents all new, modified and deleted cables to ensure the complete change history and stock continuation.

Table 1. **Selection of plants planned by computer aided planning system updating services in the course of the years**

Type of plant	**Power**	**Period of planning**	**Cables (End of planning)**	**Cables (2013)**	**Planned Cable length**	**Modified Cables**	**Removed Cables**
	[MW]		Pcs.	Pcs.	[km]	Pcs.	Pcs.
BWR69	926	1976-80	34.477	40.813	2.618	13.707	9.880
PWR	840	1974-78	20.695	30.382	1.675	24.487	9.426
PWR KONVOI	1.345	1976-82	37.706	45.894	2.889	27.865	7.104

Accordingly, plant vendors developed different programs for the design of cable systems of nuclear power plants. But often, after the construction and commissioning phases of the plants, these programs and the cable database were not improved and maintained for the entire lifetime. Consequently, not all requirements of plant operators for project work, installation and safety analysis can be complied with. Furthermore, the fast development of database software and operating systems limits the period in which a legacy system can be used. To replace the legacy system, standardized software is the optimal solution because the prospective upgrades will be taken care of by the provider of the new system.

2. Building up a New Cable Management

Nowadays, additional factors have to be taken into account in the planning and operation of cable systems of the plants. A couple of years ago, it was only necessary to provide information regarding the connection between feeders, distribution boards and components. At the time being, besides the augmented requirements for documentation, further information regarding availability and security has come to the fore. Constructors and operators of nuclear power plants need a continuous cable management system with graphical support for design, erection, operation and dismantling.

2.1 Data collection

The first step in building up a cable management system requires an evaluation of the existing documentation. The required data are usually available from the operators and are examined thoroughly before being imported.

In particular, the following documentations represent a cable plant:

- Cable data (cable number, destinations, cable type, length),
- Cable routing (description of cables path through the plant),
- Cable tray drawings and general arrangement drawings,
- Database from plant vendors or operators legacy system,
- Documentation from recent retrofits,
- Further skills and personal knowledge of operator's staff.

If the database does not provide complete cable and routing documentation, the actual situation is transferred gradually into the program if required. To complete the data, the operator can perform an evaluation and implementation of the drawings, include the cable routes in the system, draw up CAD figures of the building and of the rack plans, and track the cable routes through special methods of iteration.

2.2 Cable route network visualisation

The construction and layout plans of a system provide the details that are used for the documentation of the cable route network. The basis of the entire plant is the general layout, which provides a geometrical reference (x/y-coordinate) for each building or plant area. The plans of the individual levels of the building are subsequently transferred to the corresponding coordinates and then the constitute graphics interface as a reference file for creating a network layout.

For creating location-oriented network plans, nodes based on cable tray drawings are positioned in all rooms and places where components, breakthroughs, fire barriers or cable route branching are located. The alphanumerical element of the nodes is generated automatically on the basis of x/y-coordinates and the high node (z-coordinate) of the cable routing is defined by manual entry. The individual route channel is a segment which is represented as a connection between two nodes in the network plans. For each segment the database contains the following information:

- Length, determined by the system,
- Cable route alphanumeric,
- Number of racks,
- Type of rack/laying method,
- Voltage level,
- Redundancy.

In combination, related segments of a route section form a cable route line. The cable route line is the criterion used for the rack marking and generally facilitates the route descriptions of the individual cables. Within the graphics interface, all components are localised in the system by allocating a network node alphanumerically to each one. Room numbers and other information are stored with the so-called destination segment. For reconstructing and verifying the cable routings afterwards, the existing cable route alphanumeric (from the legacy system) can be entered into the new system.

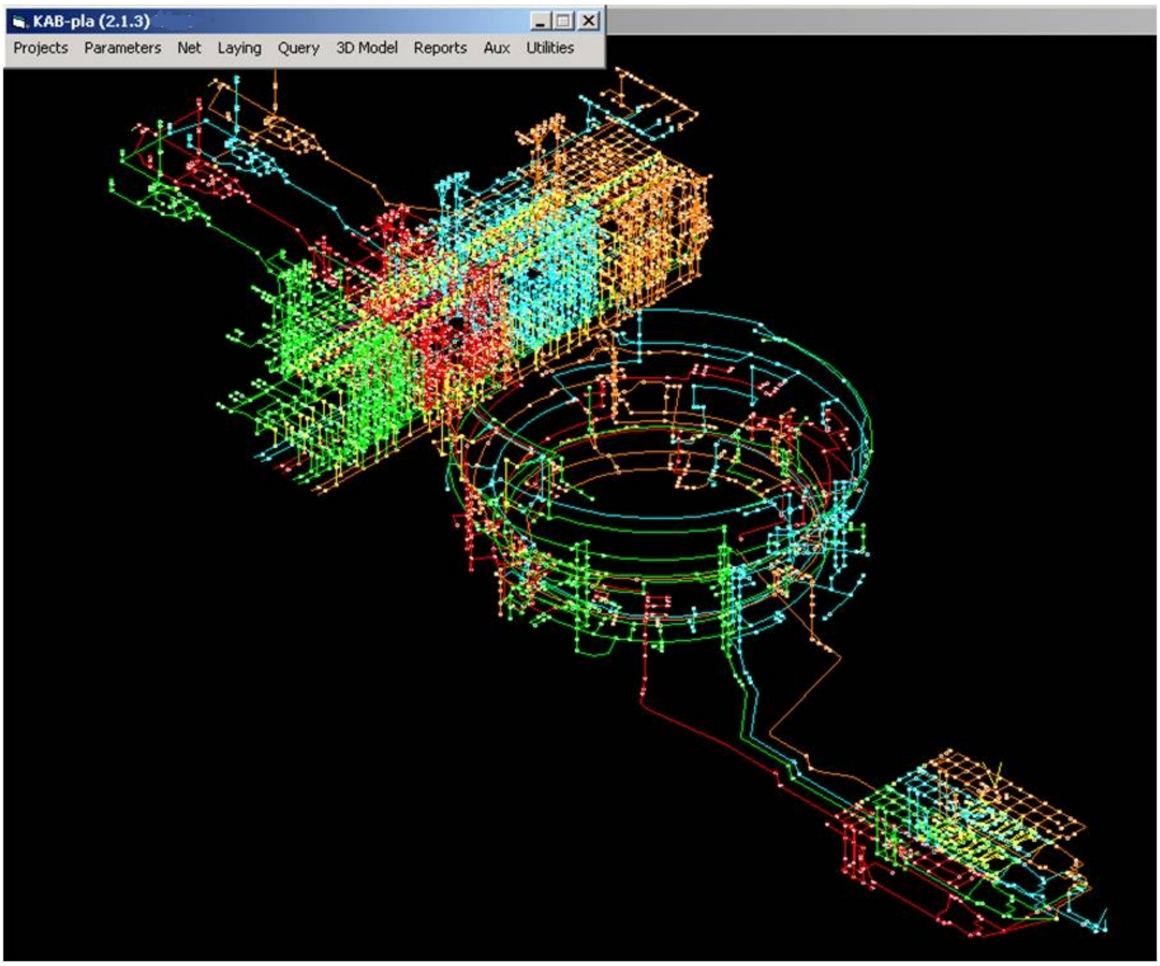


Figure 1. **Cable route network visualisation**

2.2 Network data for reconstructing and verifying the cable routing

Digitalisation of the cable routes occurs automatically in the vectoring of the route network and the localisation of the components to be cabled as described above. The data records required for the calculation are recorded directly during the transfer of network data into the processing system.

In comparison to conventional planning and documentation methods, a considerable reduction in time can be achieved by virtual cable routing on the digitalised route network. The optimum cable routing, i.e. the shortest possible cable route taking all layout criteria into account, is determined automatically for all cables.

Automatically generated cable runs require verification because in some cases there is no unambiguous cable routing available. For instance, cable routing from the legacy system may have formal errors due to the lack of system-supported logging of the cable routing. Therefore, the shortest possible cable routing established by the automatically generated cable runs does not always correspond to the actual cable run through the plant. In order to retrace the correct cable routing, known elements of the cable routing are defined based on the documentation for the cable routing, e.g. a cable laying certificate. On the basis of these aspects, the correct routing is replicated. Those cables that were dealt with according to this procedure are labelled accordingly in the database.

2.3 Continuous cable management system for the plant lifetime

After the data conversion into the operator-oriented cable management, the operator's staff shall use a continuous cable management system for all tasks and cycles of the plant's lifetime.

- Planning and Modification;
  Cable tray planning, cable planning, routing and dimensioning power cables
- Design and Construction:
  Preparation and execution of cable laying
- Operation;
  Documentation from retrofits, fire prevention and safety analysis
- Dismantling:
  Scheduling (scope) and determination of quantity (cables and cable support systems)

3. Operator-oriented Cable Management

Most cable management systems only contain features for planning and documentation. As an operating power plant has to comply with specific security and safety requirements, as a subsequence an operator-oriented cable management has to comply with additional tasks in order to cover all external and internal influences of a cable plant.

3.1 Influences for the definition of cable plans

Any cable installation in a nuclear power plant is subject to a large number of influences. Some of these that are relevant for PRA are listed below.

- Mechanical influences (protection against damage, mechanical loads, fastening, electromagnetic forces, earthquake resistance of cable trays),
- Environmental impacts (fire, temperature, radiation, oil, chemicals, humidity, gas, pressure, LOCA),
- Safety (fire behaviour, functional endurance, halogen-free insulations).

3.2 Additional tasks

During the life cycle of a nuclear power plant, the cable system and cable management system are subject to new requirements, for instance those resulting from preventive fire protection and the dismantling after the end of power generation.

- Recording of protective measures for cables against fire:
  It is possible to enter data about protective measures for specifically protecting cables (intumescent coatings, cable bandages) into the database. During the planning of the cable routing, the user is able to consider fire protective measures. With a comparison of the date of cable laying and implementation of precautionary measures, the user can detect cables outside these protective shieldings or coatings.
- Section query combined with simplified cable dimensioning:
  Before a fire protective measure (e.g. cable bandage) is installed around a cable tray,

the power cables will be checked if they have enough reserves concerning thermal heat, because the cable bandage will impair the air circulation.

- Qualified dismantlement of cables:
  To avoid having to install new cable trays, it is possible to remove cables being out of operation from the cable trays to extend the available space on the cable route network. A cable out of operation can be removed from one cable penetration to the next one. The user can consider such cables while planning new ones. Cables out of operation have to be taken into account when calculating the occupancy of cable trays and the fire loads.

4. Use for Probabilistic Fire Risk Assessment

The cable management systems of several nuclear power plants are also used for Fire PRA. In the frame of several PRA projects it was observed that fire induced cable failures can constitute the dominant contribution to fire induced core damage frequency (CDF), because cable failures can generate dangerous failure modes of components important to safety in a nuclear power plant, e.g. spurious actuation or unavailability on demand.

In the cable management system, all cables within the plant are recorded with identification number of cable, destinations, cable type, cable routing, and room numbers of the compartments passed through. Completed with the allocation data for cable number to component, a cable management system provides all necessary cable data for the analysis of the probable fire induced cable failures in the frame of Fire PRA.

4.1 Fire loads

GRS has recently performed a Fire PSA for a German BWR plant of the type BWR69 [1]. Based on the data in the cable management system, experts have quantified the fire loads from the cables. The cable management system contains the required technical data to determine the occupancy and static load on the cable trays. To calculate the fire load resulting from the cables, the technical data also include cable type related details like diameter, total weight and fire load.

The cable network data includes information on the installation type of the cable support systems and the cable routing, i.e. a description of the cable run from the origin to the endpoint. Furthermore, there is a joint cable type database for three nuclear power stations with additional details provided by the manufacturers such as data sheets and excerpts from catalogues.

It is assumed that the cables burn completely, meaning that the combustion of the cable material is at 100 %. Therefore, this approach is very conservative. Actually, when calculating the fire load, the complete fire load from cable insulation material could be reduced by a specific combustion factor for cables.

Additionally, the database contains information whether the cables are fire protected or not. For example, the cables can be protected by intumescent coatings or by fire proof encapsulations (e.g. Promat).

4.2 Processing

The processing works as follows:

- First, a room number shall be selected.
- The node data establish which nodes are located in the selected room.

- The sequence of nodes allows to identify the sections respectively node sequence of the cables in the selected room.

The segment data enable determining the length of each cable inside the selected room. The following formula is used:

$$Q = (l \times B) : 1000$$

l = length of the cable in the selected room [m]

B = specific fire load of the cable type [kWh/km]

Q = the fire load caused by the cable in the selected room [kWh]

The fire loads (Q) from each cable in the selected room are added up to establish the total fire load in the selected room.

The data in a cable management system provide fire related data for all documented cables in every room. These information, which are in principle easy to collect, are used to automatically calculate the fire load caused by the cables in the individual rooms with regard to their fire relevance. In case of fire, the possible effects caused by cables refer to the following parameters: fire load, frequency of fires according to different ignition sources and fire spread. As a result of the calculation, a list is generated which can be used for the screening in the frame of the Fire PSA. This list displays the relevant data for the selected rooms and is the basis for the data analysis.

4.3 Cable allocation

A database provides information for all cables routed through a selected room. The cables are classified according to two parameters:

- Alphanumeric value of the component(s) which are supplied via the cable, even if the cable has a different destination than the component(s).
- Cable function, depending on the information available in the cable management system.

In addition to the pure cable and cable routing data of the cable management system, fulfilling the tasks of Fire PSA [2] requires the allocation of single physical cables (cable with cable number in the cable management system) to a functional chain. Thereby, the path of energy and signals respectively can be pursued from the beginning via the branching points to their destinations.

4.4 Room-component-allocation

The cable allocations (that were generated previously) generate a room-component allocation. The plant identification code system is used to designate the components. In general, the electrical equipment of a component are not modelled separately (e.g. switches and cables of a power unit) in the fire PSA but centralised in a functional element for the failure of the component under scrutiny. Correspondingly, the individual means of function of a component are coded through the plant identification code in the database for the room-component allocation. This allows that if cables and other electrical equipment are situated in another room than the component, the component can be allocated to various rooms. The function that the cable of a component fulfils is inventorized in those rooms where the cable is laid.

5. Failure Mode and Effect Analysis

Up to now, due to limited information about cable types, structures, shielding and routing conditions, conservative assumptions had to be made about cable failure modes and their corresponding functional impact on the attached components. In a further step, GRS developed a methodology [4] based on FMEA (Failure Mode and Effect Analysis) to assess potential fire induced cable failures in a more realistic way. In addition to the cable type data stored in the cable management system, the used cable types were classified in different categories. According to experimental results [3] for fire induced cable failures from iBMB (Institute for Building Construction, and Fire Protection of Braunschweig University of Technology), four variables were assigned: PVC insulated and FRNC cable types with a second subdivision into power cables and I&C cable types.

Conclusions

The analysis of the cabling of a nuclear power plant can provide important information for sufficient modernization of cable systems, e.g. the use of advanced cable types, separation and protection issues. Detailed information about all (particularly safety related) cables in the NPP are vital for Fire PRA.

In a nuclear power plant, all departments continuously collect large quantities of data. The intelligent interlinking of individual databases can already lead to additional insights and avoid double entries. Before the actual analysis, the teams in charge of cable management and the PSA have to start in a timely manner with the checking and assessing of the relevant data for the cable plant. Furthermore, they have to take the appropriate measures to ensure the availability of the new cable management system with verified data before the start of the PSA. The evaluation of the legacy system, the implementation of the documentation, the drawing up of the cable support systems, and the inclusion of cable routings take considerable time. A cable management system can also be built step by step. Depending on priority, data of those buildings that have a high security weight can be recorded and regenerated in the first phase. The remainder of the cable plant can be addressed in the next step.

If a nuclear power plant is newly built or if substantial reconstruction takes place, the contractors are obliged to supply the operator with cable and routing data that can be further processed (no PDF files). During normal operation of a nuclear power plant, around 300 cables are laid per year. Also for those cables, the cable laying is steered through a fixed procedure, e.g. a central document such as a cable laying certificate, and the actual status is fed back into the database. With the establishment of a new cable management system that is focused on the operator, the expertise of long-time employees can be secured. Furthermore, it ensures the existence of an efficient, standardized system for the planning and documentation of a cable plant for all cycles.

References

[1] Röwekamp, M., et al. (2005) “Ausgewählte probabilistische Brandanalysen für den Leistungs- und Nichtleistungsbetrieb einer Referenzanlage mit Siedewasserreaktor älterer Bauart“, Schriftenreihe Reaktorsicherheit und Strahlenschutz, Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), BMU-2005-666, 2005

[2] Türschmann, M., J. von Linden, and M. Röwekamp (2005), “Systematisches Auswahlverfahren für probabilistische Brandanalysen“, Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), Schriftenreihe Reaktorsicherheit und Strahlenschutz, BMU-2005-667, 2005

[3] Hosser, D., O. Riese, and M. Klingenberg (2005), “Performing of Recent Real Scale Cable Fire Experiments and Presentation of the Results in the Frame of the International Collaborative Fire Modeling Project ICFMP”, und Strahlenschutz, Schriftenreihe Reaktorsicherheit Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), BMU-2005-663, 2005

[4] Frey, W., et al. (2008), „Methoden zur Abschätzung des Risikobeitrags redundanzübergreifender Brandschäden, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-A-3425, Köln, June 2008

Failure Mode and Effect Analysis of Cable Failures for Fire PSA with Quantification of Failure Mode Probabilities

Joachim Herb, Ewgenij Piljugin
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Garching, Germany

Abstract

To support Fire PSA modelling GRS has developed a methodology for performing cable failure mode and effect analyses (FMEA). In order to assist the analyst, a database has been implemented to systematically assess potential effects of cable failures caused by fire. Moreover, the different failure modes' probabilities can be systematically quantified using the correlations given by NUREG/CR-6850.

The first step of the methodology consists of grouping the cables into generic circuit types, e.g. four-wire-measurements or limit-switches.

In a second step, a FMEA of these generic circuit types is carried out to identify typical cable failure modes such as hot-shorts or fault to ground, and determine their effects on components important to safety. Thereby, affected components need not be connected directly to the damaged cable, but can be connected to it by common or branch cables.

In a third step, the failure probabilities of the cable failure modes are calculated using the correlations given by NUREG/CR-6850. These correlations require the knowledge of the number and types of circuits routed via the cable. This information can be retrieved by queries from the cable database. The electrical state of each wire within the cable can thus be determined for applying the NUREG/CR-6850 correlations.

The failure mode probabilities of different cable failures (e.g. hot-short, fault to ground) can be assigned to the corresponding probabilities of component effects (e.g. signal fails low or high). The component effects are mapped to basic events in the Fire PSA model.

Additionally, cable routing information is stored in the cable database for the entire cables of the nuclear power plant. The effect of cable failures can then be considered in the calculation of the core or fuel damage frequency by combining compartment specific fire frequencies with probabilities of failure modes and component effects respectively.

The method has been applied to the analysis of cable failures within a reference fire compartment of a German BWR plant.

1. Introduction

During the last decade, fire risk assessment has become an integral part of probabilistic safety assessments (PSA) of nuclear power plants (NPP), because past fire PSA have shown that fire induced initiating events and component failures can contribute significantly to safety hazards [1]. An important contribution to the safety hazards by fires is caused by cable failures. In [2] GRS presented a cable failure mode and effect analyses (FMEA) methodology which supports the creation of a Fire PSA model. The FMEA methodology developed by supports the PSA analyst in systematically analyzing potential malfunctions of cables and to identify potential effects of cable failures to the attached components. The cable FMEA is carried out in two phases. Generic and component and cable specific information is necessary for the two steps [2]. In [3] an enhancement for the cable FMEA methodology was presented by a quantification of the failure mode probabilities. GRS has developed also the database CaFEA (*Ca*ble *F*ailures *E*ffect *A*nalysis) to support systematically the assessment of potential effects of cable failures caused by fire [4].

In Section 2, the GRS FMEA methodology is briefly presented. In Section 3, its quantitative enhancement to quantify cable failure mode probabilities is derived. The enhanced methodology is applied in Section 4 to estimate the failure mode probabilities of all cables inside a selected fire compartment of a German BWR. Finally, this paper presents some ideas regarding the use of similar methodologies for the analysis of cable failures by external and internal events such as plant internal flooding.

2. Description of the FMEA Methodology

To assess the consequences of cable failures it is necessary that information about all cables of the entire NPP are stored in a database. The database includes information about routing of the cables, cable types (number of wires, shielding, insulation materials, etc.), components physically connected to the cable, components controlled by the cable (even if they are not physically connected to the cable), and functions of the cable (e.g. power supply, control signal, feedback signals, etc.). If multiple components are controlled by a single (multi-core) cable, this information has to be available for all those controlled components. The term *cable function* is used to identify individual circuits routed by the wires inside of such a cable.

The operating conditions of the circuits and the connected components should be included in the database, e.g. if a connected motor drive is running.

The cable FMEA can be performed for assessing the effects of cable failures on the connected components and corresponding circuits after all information has been consolidated in the database. Then the FMEA results can be used for the modelling process of the PSA.

2.1 Generic and specific FMEA methodology

In the GRS FMEA methodology two steps are used to improve comprehensibility and completeness. In the first step, a generic FMEA is performed for all relevant electrical circuit types.

In a NPP safety systems and therefore also their cables are implemented several times to fulfill the construction principle of redundancy. Therefore, a large number of circuits can be grouped into generic circuit types with identical layout. This allows reducing the amount of analytical work to be performed by the FMEA experts.

Generic circuit types are based on information about the type of the connected components (e.g. drive, sensor, switchgear) and their operating conditions, and the signal type (e.g. feedback signal, control signal, power supply). The potential cable failures are identified (e.g. hot-short, short-to-ground, open circuit) and then their effects are determined on the affected components (e.g. signal too low/high, detected signal failure, loss of power supply). These generic FMEA results are stored in the database CaFEA.

The working steps of the generic FMEA are represented in Figure 1 [2].

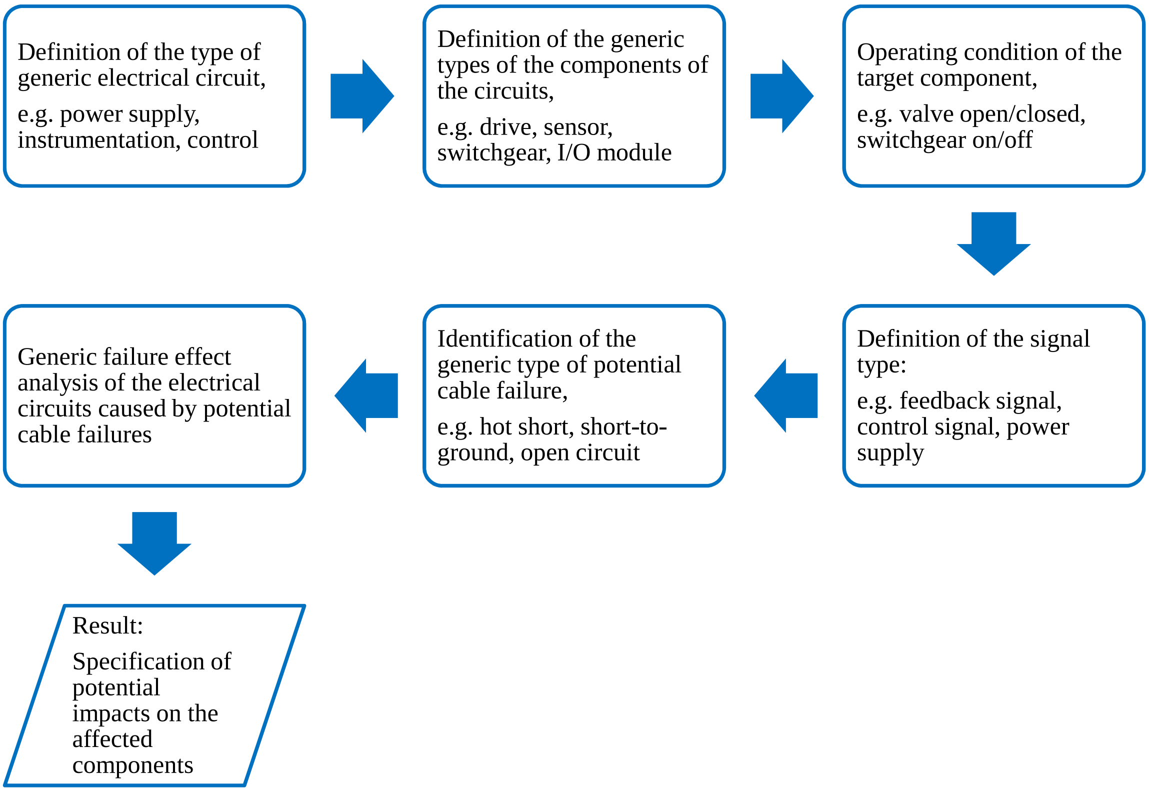


Figure 1. **Generic phase of CaFEA, from [2]**

The relevant generic circuit types are defined by screening the list of safety related components (e.g. provided by a Level 1 PSA for full power plant operational states). Circuit types found to be relevant are power supply circuits, instrumentation circuits or control circuits. For each circuit type "source" and "target" component types have to be identified. Examples of source component types are switchgear, electronic board, and relay. Typical target component types are pump, valve, motor drive, and measurement sensor.

Additionally, it is necessary to specify a sub-type or signaling type for both, the source and the target components. This signaling type is needed to distinguish between circuits types connected to one component (type). E.g. circuits of type "power supply" as well as "feedback signal" might be attached to a valve. For the circuit type "power supply" the source component sub-type might be "power supply" and the target component sub-type "motor". For the circuit type "feedback signal" the source component sub-type might be "drive control module" and the target component sub-type "control head".

The operating condition of the target component (type) influences the possible effects on the attached component. Therefore, the generic FMEA has to be performed for all operating conditions of the generic circuit type.

2.2. Component and compartment specific FMEA

After the generic FMEA the cable and component specific FMEA has to be carried out (see Figure 2, [2]). The specific conditions of the cables are considered in this step (e.g. operational mode, construction of the cable, etc.).

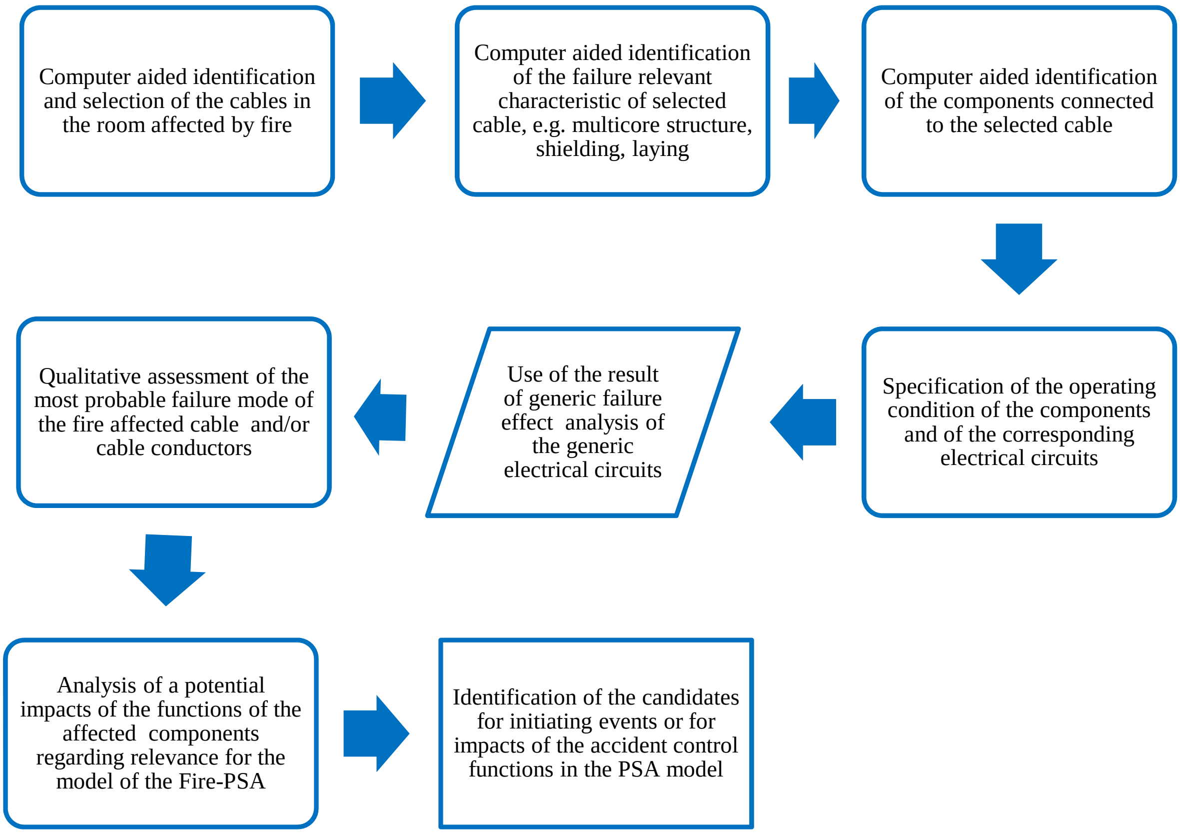


Figure 2. **Room specific phase of CaFEA, from [2]**

Using the database CaFEA all cables within a fire affected compartment can be identified. CaFEA contains information about the cable routing for all cables of the entire NPP provided by the licensee. CaFEA provides to the FMEA expert those characteristics of the identified cables relevant for the FMEA, e.g. the number of conductors, shielding, routing, the type of insulation.

Also, CaFEA identifies the components connected to each cable. A “start” as well as an “end” component is defined for each cable. These are the components connected physically to the cable. The “target” component is the component controlled by the cable or which provides data to be transmitted by the cable. For the start, the end and the target component the component types and sub-types have to specify by the FMEA expert. The effects of the fire induced cable failure are analyzed for the target component.

The operating condition of the target component has to be specified. It is based on the plant operating manual and the safety specification of the nuclear power plant. The signal type (circuit type) has to be specified by the FMEA expert. CaFEA compares the specific information of a cable and the attached components to the generic FMEA results. The component effects determined by CaFEA have to be accepted or corrected by the FMEA expert as results for the cable specific FMEA.

3. Quantitative Enhancement

The probabilities of the component effects determined by the cable FMEA have to be specified, so that they can be use within a Fire PSA. This requires the knowledge of the probabilities of the different individual cable failure modes. The analysis is based on the assumption that that hot-shorts between wires are only possibly inside a single cable and not between wires of neighboring cables. Based on the type and number of the generic cable functions within each cable the probability of the different cable failure modes can be derived.

It is assumed that the only possible cable failure mode for the power supply cable function is a short-to-ground resulting in a (detected) failure of the attached component.

An analysis is required for cables with feedback or control signals, which has to be based on the actual circuit layout. Results derived from experiments [5], [6] provide probabilities for each failure mode (see equations (1) and (2)).

$$P_{FM} = CF \cdot P_{CC} \tag{1}$$

$$\mathrm{P_{CC}} = \frac{\mathrm{C_{Tot}} - \mathrm{C_G}}{(\mathrm{C_{Tot}} - \mathrm{C_G}) + 2\mathrm{C_G} + 3}, \ \mathrm{CF} = \frac{\mathrm{C_T}\left(\mathrm{C_S} + \left(\frac{1}{2\mathrm{C_{Tot}}}\right)\right)}{\mathrm{C_{Tot}}} \tag{2}$$

c_{Tot} is the total number of wires within a cable, c_G the number of ground wires, c_S the number of source wires, and c_T the number of target wires. P_{FM} is the failure mode probability, P_{CC} the probability that a conductor-to-conductor short occurs before a ground short and CF a configuration factor. The equations are valid for shielded cables (like all I&C cables).

An example for such cable is the one shown in Figure 3. It connects two 4-wire measurement transmitters with two analogous input modules of a signal acquisition unit. A measurement signal (e.g. a pressure measurement) is transformed into an electrical current signal (from 0 mA or 4 mA to 20 mA) by the 4-wire measurement transmitters. The analogous input module records this current signal by measuring the voltage across a resistor. All ground levels and also one of the signal wires (*I1*) are on the same electrical (ground) potential.

Seven different cable failure modes can be identified for this circuit (see Table 1) based on shorts between the different wires of the cable.

Due to the common ground potential, shorts to one of the wires *I1* or *M* in either of both circuits within the cable result in the same effect. The cable failure short-to-ground between *La+* and *M* can either be short-to-ground with *Ma* or *Mb* in Figure 3.

For the first (upper) circuit, a “signal too high” effect can occur, if the wire *I2* is connected to *L+* of both circuits (*La+*, *Lb+*). So the number of target wires is one and the number of source wires is two.

This can be generalized to *N* circuits within a cable instead of two. Then there are *N* source wires. The wires *I1* and *M* are on electrical ground potential resulting in a detected failure of the circuit eventually, so the number of ground wires is 2**N*.

Using $N = 2$ and applying equations (1) and (2) this results in a probability of about 7 % for “signal too high”.

The effect “signal too low” can result from a short between *La+* and one of 2**N* = 4 source wires (*I1a*, *I1b*, *Ma*, *Mb*). The probability calculated by equations (1) and (2) is about 23 %.

Correspondingly, the probability for the effect "signal undefined" is about 5 % (there is one target wire (*I2a*) and *N* - 1 = 1 source wires (*I2b*)).

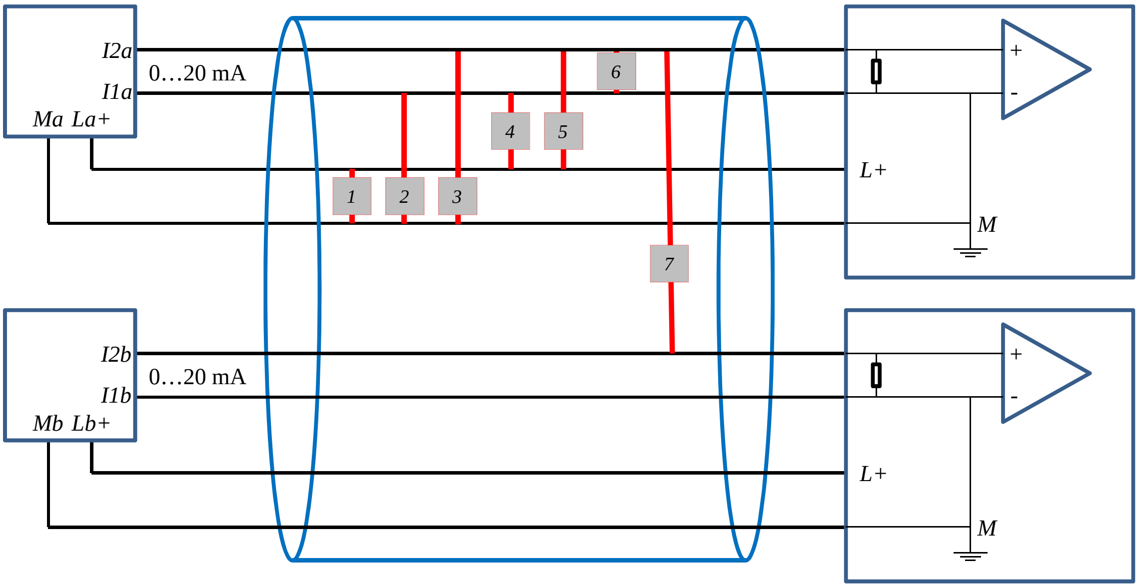


Figure 3. **Example for a generic circuit FMEA: Two 4-wire measurement transmitters connected by a single cable and intra-cable shorts with different failure modes, from [3]**

This analysis has to be repeated for all generic circuit type and should consider that a certain circuit type is embedded *N* times within a single cable.

Table 1. **Ground- and hot-shorts of the 4-wire measurement transmitter circuit (in Figure 1), the cable failure modes and the resulting component effects**

Short #	Cable failure mode	Component effect
1	Short-to-ground (*L+* - *M*) (of power supply)	(Detected) failure of measurement
2	No failure, because cables are both on ground potential (*I1* – *M*)	No failure
3	Short-to-ground (*I2* – *M*)	Signal too low (possibly also detected failure)
4	Short-to-ground (*I1* – *L+*) (of power supply)	(Detected) failure of measurement
5	Hot short (*I2* – *L+*)	Signal too high
6	Short-to-ground (*I2* – *I1*)	Signal too low (possibly also detected failure)
7	Hot short (*I2a* – *I2b*)	Signals undefined (one too high, the other too low)

4. Reference Application

The methodology was applied to all cables within a certain fire compartment of a German BWR to demonstrate its applicability [3]. This compartment contains 432 cables with together 932 different cable functions.

In a first step the generic cable types of these cables were determined. Thereby, it was found, that two of the 432 cables contained different cable functions (temperature measurement and 4-wired pressure sensors). These cables would require a special (generic) FMEA treatment. To simplify the analysis, they were neglected. Nine different generic circuit types were identified for the remaining 430 cables with 923 cable functions (see Table 2). The resulting component effects for these generic circuit types are also given in Table 2. They were the result of the generic FMEA.

Table 2. **Generic circuit types, their components effects and their fraction of all cable functions in reference compartment form [3]**

Generic circuit types	Effects on Component	Fraction
AC power supply (220 V, 400 V)	Detected failure	42,8%
DC power supply (220 V, 24 V)	Detected failure	3,7%
DC power supply for measurement transmitters (24 V)	Detected failure	0,9%
4-wire measurement transmitter	Detected failure, signal too low, signal too high, undefined signal failure	5,3%
3-wire measurement transmitter	Detected failure, signal too low, signal too high, undefined signal failure	0,7%
Temperature measurement sensor	Signal too low	1,7%
Common cable for temperature measurement sensors	Signal too low	6,7%
Dual-position indicator of motor driven devices	Detected failure	20,6%
Binary position indicator	Detected failure	17,7%

As described in section 3 it was assumed that due to the protection concept of the electrical circuits of German NPP the three different power supply circuit types always fail by short-to-ground resulting in a detected failure. Consequently, the probability of a detected failure is 100 %.

For position indicator circuit types the analysis demonstrated that they can only result in detected failure because of the automatic plausibility check of the data acquisition units. In such a position indicator circuit a ground-short would result in the detected loss of an input signal, a hot-short would result in the simultaneous indication of an open and close position which would also be detected as a failure of the signal.

For the circuit types of 3- and 4-wire measurement transmitters different failure modes were identified. For the circuits used for temperature measurements only the effect “signal too low” was found.

The failure effect probabilities for the 49 circuits within 16 cables in the reference compartment connected to 4-wire measurement transmitters are shown in Figure 4. The different probabilities result from the different numbers of circuits within each cable.

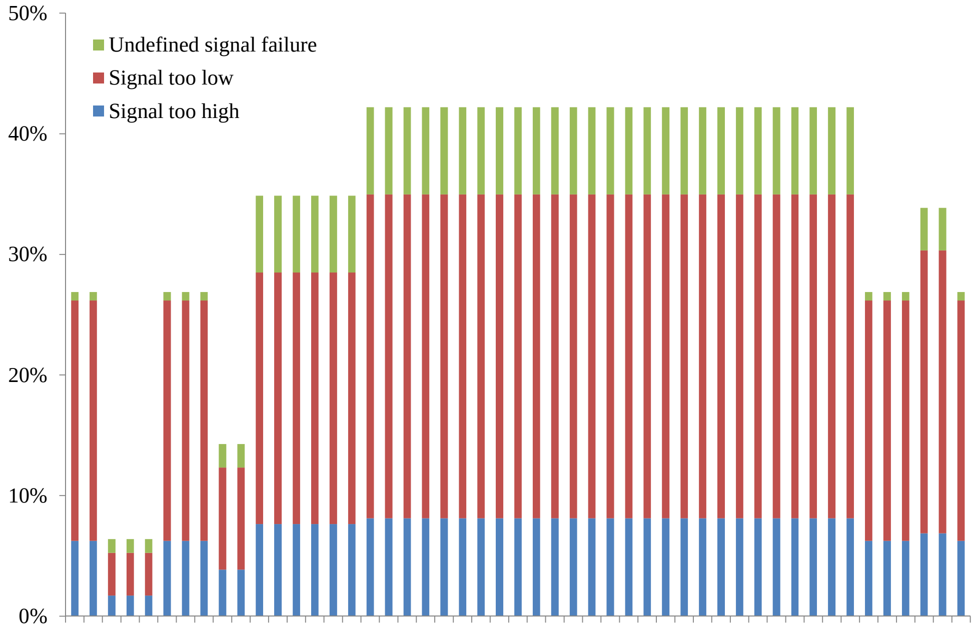


Figure 4. **Probabilities of the undetected failure effects of the 4-wire measurement transmitter circuits inside the reference room, from [3]**

5. Conclusion

GRS has developed a methodology for applying a FMEA to cables within a nuclear power plant for Fire PSA, based on a two-step approach first carrying out a generic and then a cable specific FMEA. Also included is an approach to quantify the cable failure mode probabilities.

The method proved applicable due to the reduction of the work load for the analyst. For the approximately 1000 cable functions in a reference compartment of a NPP only nine generic circuit analyses had to be performed. By these nine cases to be analyzed all relevant circuit failures modes and the corresponding effects of the connected components could be identified and their probabilities determined.

No control cables connecting e.g., the control room with switchgear equipment had to be analyzed, because the selected reference compartment was located in the reactor building and does not comprise those types of cables. Also, no (direct) spurious actuations of components (e.g. of valves or of motor drives) were identified. For the switchgear building these types of component effects have to be expected in a similar analysis. Nevertheless, the GRS FMEA methodology should also be applicable to cables connecting e.g. the reactor protection system with switchgear equipment.

The quantification is based on the correlations given in [5] for cables typically used in U.S. NPP. These correlations should be verified by experiments carried out specifically for cables of the NPP for which the Fire PSA is performed.

To increase the efficiency of the methodology the calculation of the individual cable failure mode probabilities can be automated in a database application such as the GRS CaFEA database after a generic circuit analysis has been performed.

Cable failures are also relevant for other probabilistic hazard analyses, such as for plant internal or external flooding. The approach of the presented cable FMEA can be applied principally for all kinds of hazards analyses associated with damage of the cables. Adaptions would be necessary for the information stored in the database (e.g. failure criteria for shorts by flooding) and correlations used for quantifying the failure probabilities.

Acknowledgements

This work has been sponsored by the Federal German Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) within the frame of the project 3611R01302.

References

[1] H.-P. Berg, M. Röwekamp (2010), Current Status of Fire Risk Assessment for Nuclear Power Plants. Nuclear Power (Ed. P. Tsevtkov), Sciyo, 2010

[2] J. Herb, E. Piljugin (2011), Failure Mode and Effect Analysis of Cable Failures in the Context of a Fire PSA, Proceedings of ANS PSA 2011 International Topical Meeting on Probabilistic Safety Assessment and Analysis, Wilmington, NC, USA, 2011

[3] J. Herb, E. Piljugin (2014), Enhancement of a Cable Failure Mode and Effect Analysis (Methodology) for Fire PSA by Quantification of Failure Mode Probabilities. Nordic PSA Conference – Castle Meeting 2013, Stockholm, Sweden, 2013

[4] J. Herb, E. Piljugin (2008), FMEA of Cable Failures within Fire PSA. Proceedings of the 9th International Probabilistic Safety Assessment and Management Conference (PSAM 9), Hong Kong, China, 2008

[5] Electric Power Research Institute / Nuclear Regulatory Commission Office of Research (EPRI/NRC-RES) (2005), Fire PRA Methodology for Nuclear Power Facilities Volume 2, EPRI 1011989, NUREG/CR-6850 Final Report, Palo Alto, CA, USA, 2005

[6] Electric Power Research Institute (EPRI) (2002), Spurious Actuation of Electrical Circuits Due to Cable Fires: Results of an Expert Elicitation, EPRI 1006961, Palo Alto, CA, USA, 2002

Extension of the Plant Specific Database of Fire Protection Systems and Components

Burkhard Forell, Svante Einarsson
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany

Abstract

In the frame of state-of-the-art Fire PSA fire event trees as well as fault trees have to be established considering plant specific characteristics and boundary conditions for the scenarios to be investigated. This enables the analyst to successfully estimate the corresponding branch point probabilities and end states, typically to establish fire induced core damage frequencies or fuel damage frequencies.

In the context of modelling plant specific fire event trees, reliability data with regard to fire protections means are required. In the most recent version of the technical document on PSA data supplementing the German PSA Guide as published in 2005, data to be applied within the fire event tree analysis, in particular technical reliability data for various active fire protection features, are presented. The reliability data have been provided resulting from plant specific assessments of operating experience from different nuclear power plants. In order to update the already existing reliability data (in particular, failure rates per hours of plant operation) as well as extending the database to include an additional plant unit the following components and systems have been analyzed in six nuclear power plant units at five sites in Germany:

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

Plant specific reliability data have been gained by analyzing the documentation of periodic in-service inspections as well as additional information and reports resulting from findings. For more complex systems such as the fire detection system, fault trees are provided for estimating the reliability of these systems in addition to the reliability of the individual components. The existing database has been recently extended to cover approximately 111 plant operational years of six units of different types of nuclear power plants in Germany.

1. Introduction

A variety of data is needed for performing Fire PSA and quantifying the fire specific event trees with corresponding branch point probabilities and end states for core damage. These data include fire occurrence frequencies, fire spreading parameters, unavailability of active and passive fire protection features, and failure rates for human actions in case of fire.

Full scope Level 1 Probabilistic Safety Analyses (PSA) is mandatory to be performed for all plant operational states including plant internal fires (Fire PSA) in the frame of the Periodic Safety Reviews to be obligatory to be carried out every ten years for nuclear power plants (NPP) in Germany as required by the Atomic Energy Act and according to the corresponding Guide. Moreover, the German high level “Safety Requirements for Nuclear Power Plants” [1] require probabilistic assessment in case of any plant modification which might change the core or fuel damage frequency.

The high level regulatory documents are supplemented by technical documents on PSA methods [2] and data [3] to provide detailed guidance for the PSA analyst. The latter also covers data for Fire PSA, in particular tables with generic reliability data for active fire protection features.

In order to model the plant specific fire event trees in an as far as possible realistic manner, plant specific failure rates and resulting unavailabilities per demand for fire protection features are needed. In the past, such data have been collected within different projects [4], [5], and [6]. In addition, these data were extended and improved within a recently finished project [7]. Meanwhile 111 reactor years from six German NPP units are covered (cf. Table 1). More specific, the database meanwhile consists of four BWR units with approx. 69 plant operational years (before the extension, data of three BWR units with 23 plant operational years were available) and three PWR units with approx. 42 plant operational years (before the extension, data from only 17 plant operational years in the same plants were available). These data are being included in the guidance document on PSA data and are applicable within the PSA required by the recent German “Safety Requirements for Nuclear Power Plants” [1] to be conducted either in the frame of the last remaining (Periodic) Safety Reviews for German plants or in case of plant modifications potentially affecting PSA results.

Table 1. **List of NPP units investigated**

No.	Reactor Type	Actual Investigation Period	Former Investigation Period	Reference
1	Konvoi PWR	1998 – 2012 (15 a)	1994 – 1997 (4 a)	[5]
3	BWR-72	2000 – 2012 (13 a)	1988 – 1994 (7 a)	[4]
4	Pre-Konvoi PWR	1995 – 2010 (16 a)	1988 – 1994 (7 a)	[4
5	BWR-69	2002 – 2011 (10 a)	1993 – 2001 (9 a)	[6]
6	BWR-69	2001 – 2010 (10 a)	not investigated	-

The systems and equipment analyzed are:

- Fire detection systems consisting of the main fire alarm panel and subsidiary alarm boards, detection drawers inside such boards, detection lines, and the automatic fire detectors or manual fire alarm buttons connected to the lines,
- Fire dampers in the ventilation systems,
- Smoke extraction dampers in ventilation ducts and smoke vents in roofs and walls,
- Fire doors, partly equipped with devices to keep them in open position (hold-open devices),
- Fire fighting systems and equipment such as water pumps, remote controlled valve stations of water deluge systems, and hydrants (field hydrants, wall hydrants, and foam wall hydrants).

2. Approach for Generating Technical Reliability Data for Active Fire Protection Features

The technical reliability data are derived from results of in-service inspections carried out for active fire protection features as present in the NPP units to be considered in the analysis. The active function of all systems and components in German NPP is inspected regularly via component specific inspection programs. The observed findings of these inspections, i.e. anticipated functional deteriorations and failures are documented in the inspection records. In addition to the records of periodic in-service inspections (including the inspection procedures), the resulting reports on findings and deviations, maintenance orders, and repair reports were analyzed. Based on these documents and additional consultation of the plant personnel involved in inspection and maintenance of the corresponding fire protection features, each finding of an in-service inspection has been analyzed regarding its required function in case of fire. In this context, it has to be distinguished between findings representing only a deficiency not inadmissibly deteriorating the required fire protection function of the component and those findings representing a failure of the required function. The latter are those findings accounted for in the statistical analysis for generating reliability data.

Self-signaling deficiencies or failures observed independently of an in-service inspection are not accounted for in the analysis. It is known from the operation experience that plant operators usually do react on self-signaled failures by taking compensational measures and by carrying out repair work at these components within a short time period.

For a consistent assessment of the raw data suitable definitions of “failure” and “deficiency” are needed to receive realistic values for failure rates to be applied in Fire PSA. Considering the relevance of the affected components or systems in the event trees, a careful assessment by engineering judgment based on expert knowledge is needed for determining if the documented findings can be interpreted as functional failures (“unavailability”) or as deficiencies only. In this context, detailed knowledge of the plant specific boundary conditions is evident for the assessment. This requires a thorough plant walk-through for all plant areas where the fire protection features to be investigated are in place. In addition, for a meaningful and consistent assessment close co-operation with the plant personnel in charge of inspections and maintenance of the different systems and components is necessary.

In addition to the number of failures k, the number of components with the same design characteristics, the cumulated observation period T and the inspection interval of the components are collected. In the past (cf. [4] to [7]), failure rate as well as unavailability per demand were calculated from the raw data for active fire protection features. However, the most recent reliability data cover failure rates per hours of plant operation $\lambda(t)$ only, because it is assumed that the failure characteristics of the fire protection features correlates well with the time. The expected value of the failure rate becomes

$$E(\lambda)=\frac{k+0.5}{T}$$

For generating generic data it has to be decided whether the components of different reference plants can be considered as almost similar with comparable inspection practice. Then the components are binned together in one common data pool and treated as if belonging to the same plant. For most of the features analyzed the components have not been pooled, because individual components from different suppliers are installed in each plant, within individual operating history, maintenance strategy, etc. For determining generic data for such components, plant specific reliability values are calculated by a superpopulation approach [8]. This approach considers the differences between components from different plants which increases the uncertainties of the estimated failure rates.

Data sets of electric and electronic equipment being pre-manufactured externally by the same manufacturer and installed in different plants are pooled together for generating a generic data set. This procedure is applied to fire detection system equipment with the exception of aspirating smoke detectors (ASD) and manual call points. The distribution ranges of the estimated values are computed by a statistical estimation based on the approach of Bayes. Two steps are performed, first a gamma distribution is generated via a Bayes non-informative approach; in a second step, an algorithm [9], [10] is applied for considering additional sources of epistemic uncertainties. The result is a non-parametric distribution of the estimates with the relevant quantiles, mean value, and standard deviation.

In-service inspections do not address system design; therefore the data collected do not reflect possible design failures. Examples for design failures might be a fire door with insufficient fire resistance rating or a smoke detector located too close to an air inlet vent.

3. Specific Technical Reliability Data Recently Generated for Different German Nuclear Power Plants

The raw data for determining failure rates have been collected from six reactor units of different type and age. The fire protection features with active functions being investigated are sub-divided into

- automatic fire detection systems (Table 2),
- fire dampers, smoke vents, and fire doors (Table 3), and
- fire extinguishing systems and equipment (Table 4).

Each table contains the component/system name and type, the number of components observed, their associated test interval, the cumulated observation periods of all components, the observed number of failures, and the resulting failure rate with their 5 %, 50 % and 95 % quantiles, their mean value and corresponding standard deviation. The results of the data collection and statistical processing are explained in detail in the following paragraphs.

3.1 Automatic fire detection systems

An automatic fire detection system contains several components, such as fire alarm panels and boards, fire detection drawers, detection lines and detectors of different type. The functional structure is presented by means of an exemplary fault tree in Figure 1. The faults tree covers technical failures and is representative for an automatic fire detection system of a German NPP. With the data presented in Table 2 it is possible to calculate the top event “no fire alarm indication” of the fault tree for each of the reference plants. On the bottom of the tree the state that one or more fire detectors in one fire compartment do fail is portrayed. The fire detectors give a signal being transmitted via the fire detection lines. If the fire detection line or all its associated fire detectors in the fire compartment fail, there will be no further transmission of the fire detection signal. However, there may be other fire detectors installed in the same fire compartment not connected to the same fire detection line. In case of a failure of all detection lines or of their connected fire detectors the fire detector signal will not be transmitted to the detection drawer. Another possibility of a non-successful signal transmission is a detection drawer malfunction. Similar to the detection lines, there may be several detection drawers transmitting an alarm signal from the same fire compartment. The signal may be transmitted via the detection drawers α directly to the main fire alarm panel or being further processed through a subsidiary fire alarm board. Similarly, the signal may be transmitted via different detection drawers α, β, γ, etc. and additionally through one or several subsidiary fire alarm boards. If one of these boards or the detection drawers α, β, γ, etc. fail (if all present), the signal is discontinued.

Four generations of fire detection systems components are to be distinguished:

- The first generation of fire detection systems, meanwhile taken out of operation, was based on analog technique,
- the second generation was based on digital technique newly introduced at that time,
- fire detection systems of the third generation of were installed from the mid-1990es using more powerful processors than those of the second generation, and
- a fourth generation of fire detection systems is meanwhile available on the market, however, there is up to now no operating experience available from nuclear power plants in Germany.

Table 2 provides results for components, in particular of the second and third generation. For many components of the fire detection system no failures have been observed in the frame of in-service inspections. The main reason is that the reliability of electric and electronic equipment is higher than that of mechanical equipment such as dampers, etc. Moreover, most types of failures of the fire detection components are self-signaling ones having been observed independently of in-service inspections. Therefore, they do not have to be interpreted as unavailability.

The ionizing detectors installed in the plant units being investigated are of the second generation, type A and B. For both types of detectors functional failures have been identified, although most plants have not recorded ionizing detector failures. Findings with the comment “malfunctioning detector” have been interpreted as failure. However, observations such as “detector mounting broken, detector functioning” or “LED on detector malfunctioning” have been interpreted as deficiencies only.

Large amounts of optical smoke detectors are in place in all of the plants under consideration. Only in one plant, failures not self-signaling have been recorded in the frame of in-service inspections. In this plant, 11 findings have been interpreted as failures for a total of 7600 detectors over a time period of 7 years. However, if not taking into account the few (5) detectors that were installed in restricted areas only 6 failures haven been recorded. The reason for this phenomenon is the high radiation level in the restricted area. Meanwhile an electronic reducer card has been implemented in the detectors able to manage this problem.

No findings to be interpreted as failures of the required function were observed for heat detectors, IR-flame detectors, multi-criteria detectors and manual call-points in the operating experience of the entire plant units considered during the observation period.

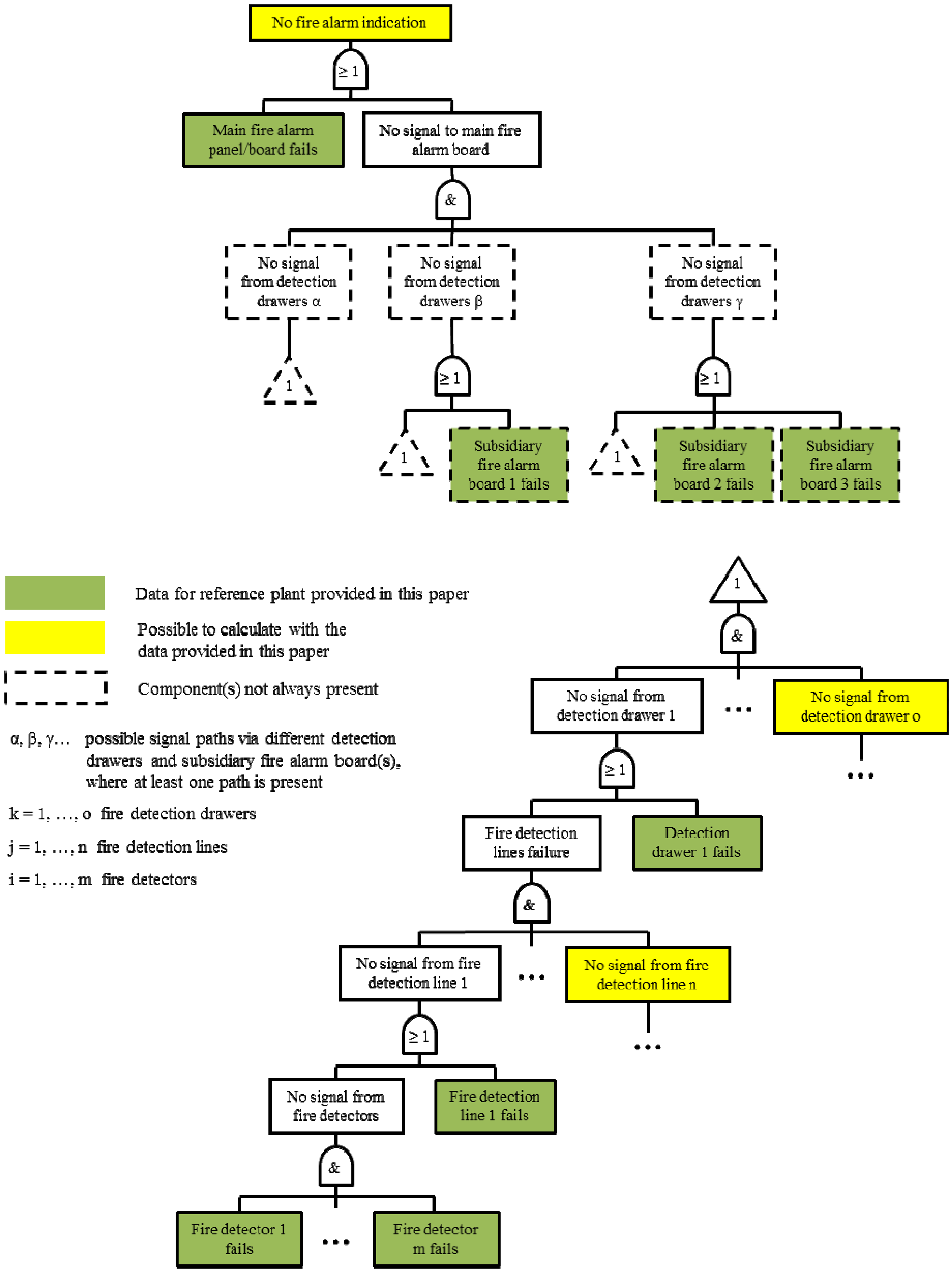


Figure 1. **Exemplary fault tree for technical failures of a fire detection system [11]**

With respect to aspiration smoke detectors (ASD), a major development resulted from the replacement of the specifications of the Comité Européen des Assurances (now: Insurance Europe) [12] (for generation 1) by the European Standard EN 54 Part 20 [13] (for generation 2). With the new specification, changes in the aspirated volume flow of ± 20 % need to be recognized by an ASD, whereas the old systems recognized changes of ± 50 %. Flow changes occur due to clogging of the inlets or due to leaks in the pipework of an ASD and are the primary cause of failures.

Table 2. **Failure rates of components of the fire detection system**

Plant	Active Fire Protection Systems and Components	Number of components	Test interval [years]	Observation period [h]	Number of failures	Failure rate [1/h] 5 % quantile	50 % quantile	95 % quantile	Mean value	Standard deviation
1	- Main fire alarm panels (Gen. II)	4	1/4	438 157	0	2.64E-09	3.59E-07	4.89E-06	1.14E-06	2.19E-06
4	- Main fire alarm panels (Gen. II)	1	1/4	103 032	0	1.12E-08	1.53E-06	2.08E-05	4.86E-06	9.32E-06
5	- Main fire alarm panels (Gen. II)	2	1/4	333 072	0	3.47E-09	4.72E-07	6.44E-06	1.50E-06	2.88E-06
6	- Main fire alarm panels (Gen. II)	2	1/4	175 320	0	6.59E-09	8.98E-07	1.22E-05	2.86E-06	5.47E-06
1	- Main fire alarm panels (Gen. III)	4	1/4	263 136	0	4.39E-09	5.98E-07	8.15E-06	1.90E-06	3.65E-06
3	- Main fire alarm panels (Gen. III)	2	1/4	154 752	0	7.46E-09	1.02E-06	1.39E-05	3.24E-06	6.20E-06
6	- Main fire alarm panels (Gen. III)	1	1/4	87 648	0	1.32E-08	1.80E-06	2.45E-05	5.71E-06	1.10E-05
1	- Subsidiary alarm boards (Gen. II)	13	1/4	1 275 024	0	9.06E-10	1.23E-07	1.68E-06	3.93E-07	7.53E-07
4	- Subsidiary alarm boards (Gen. II)	10	1/4	1 003 920	0	1.15E-09	1.57E-07	2.14E-06	4.99E-07	9.56E-07
5	- Subsidiary alarm boards (Gen. II)	5	1/4	911 568	0	1.27E-09	1.73E-07	2.35E-06	5.49E-07	1.05E-06
6	- Subsidiary alarm boards (Gen. II)	11	1/4	999 324	0	1.16E-09	1.58E-07	2.15E-06	5.01E-07	9.60E-07
1	- Subsidiary alarm boards (Gen. III)	17	1/4	1 118 328	0	1.03E-09	1.41E-07	1.92E-06	4.48E-07	8.58E-07
3	- Subsidiary alarm boards (Gen. III)	24	1/4	1 454 400	0	7.94E-10	1.08E-07	1.47E-06	3.44E-07	6.60E-07
6	- Subsidiary alarm boards (Gen. III)	3	1/4	116 150	0	9.94E-09	1.36E-06	1.85E-05	4.31E-06	8.26E-06
1	- Detection drawers (Gen. II)	49	1/4	4 938 024	0	2.34E-10	3.19E-08	4.34E-07	1.01E-07	1.94E-07
3	- Detection drawers (Gen. II)	36	1/4	3 838 752	0	3.01E-10	4.10E-08	5.58E-07	1.30E-07	2.50E-07
4	- Detection drawers (Gen. II)	142	1/4	14 260 944	0	8.10E-11	1.10E-08	1.50E-07	3.51E-08	6.73E-08
5	- Detection drawers (Gen. II)	37	1/4	6 091 704	0	1.90E-10	2.58E-08	3.52E-07	8.22E-08	1.58E-07
6	- Detection drawers (Gen. II)	43	1/4	3 769 380	0	3.06E-10	4.17E-08	5.69E-07	1.33E-07	2.55E-07
1	- Detection drawers (Gen. III)	69	1/4	999 432	0	2.55E-10	3.47E-08	4.72E-07	1.10E-07	2.11E-07
3	- Detection drawers (Gen. III)	163	1/4	9 877 800	0	1.17E-10	1.59E-08	2.17E-07	5.07E-08	9.72E-08
6	- Detection drawers (Gen. III)	8	1/4	322 151	0	3.59E-09	4.88E-07	6.65E-06	1.55E-06	2.98E-06
1	- Detection lines (Gen. II)	860	1/4[a]	86 636 689	0	1.33E-11	1.82E-09	2.47E-08	5.78E-09	1.11E-08
3	- Detection lines (Gen. II)	90	1/4[a]	9 596 880	0	1.20E-10	1.64E-08	2.23E-07	5.22E-08	1.00E-07
4	- Detection lines (Gen. II)	554	1/4[a]	54 619 968	0	2.12E-11	2.88E-09	3.92E-08	9.17E-09	1.76E-08
5	- Detection lines (Gen. II)	519	1/4[a]	86 414 808	0	1.34E-11	1.82E-09	2.48E-08	5.80E-09	1.11E-08
6	- Detection lines (Gen. II)	479	1/4[a]	41 989 140	0	2.75E-11	3.75E-09	5.10E-08	1.19E-08	2.29E-08
1	- Detection lines (Gen. III)	1046	1/4[a]	68 810 064	0	1.68E-11	2.29E-09	3.12E-08	7.28E-09	1.40E-08
3	- Detection lines (Gen. III)	1800	1/4[a]	110 086 560	0	1.05E-11	1.43E-09	1.95E-08	4.55E-09	8.72E-09
6	- Detection lines (Gen. III)	204	1/4[a]	891 941	0	1.30E-09	1.76E-07	2.40E-06	5.61E-07	1.08E-06
	- Automatic fire detectors									
1	- Ionization smoke detectors (Gen. II, Type A)	2196	1[a]	365 703 893	0	3.16E-12	4.30E-10	5.86E-09	1.37E-09	2.62E-09
3	- Ionization smoke detectors (Gen. II, Type A)	300	1[a]	31 989 600	0	3.61E-11	4.92E-09	6.70E-08	1.57E-08	3.00E-08
5	- Ionization smoke detectors (Gen. II, Type A)	963	1[a]	84 410 410	2	2.43E-09	1.94E-08	9.13E-08	2.96E-08	3.21E-08
6	- Ionization smoke detectors (Gen. II, Type A)	966	1[a]	84 644 496	0	1.37E-11	1.86E-09	2.53E-08	5.92E-09	1.13E-08
1	- Ionization smoke detectors (Gen. II, Type B)	16	1[a]	2 127 605	0	5.43E-10	7.40E-08	1.01E-06	2.35E-07	4.51E-07
5	- Ionization smoke detectors (Gen. II, Type B)	459	1[a]	40 258 342	1	1.89E-09	2.16E-08	1.26E-07	3.73E-08	4.70E-08
6	- Ionization smoke detectors (Gen. II, Type B)	235	1[a]	23 755 860	1	3.21E-09	3.66E-08	2.13E-07	6.32E-08	7.97E-08
1	- Optical smoke detectors (Gen. II, Type A)	1114	1[a]	185 627 197	0	6.22E-12	8.48E-10	1.16E-08	2.70E-09	5.17E-09
3	- Optical smoke detectors (Gen. II, Type A)	300	1[a]	31 989 600	0	3.61E-11	4.92E-09	6.70E-08	1.57E-08	3.00E-08
4	- Optical smoke detectors (Gen. II, Type A)	2444	1[a]	214 270 368	0	5.39E-12	7.34E-10	1.00E-08	2.34E-09	4.48E-09
5	- Optical smoke detectors (Gen. II, Type A)	406	1[a]	35 585 088	0	3.25E-11	4.42E-09	6.02E-08	1.41E-08	2.70E-08
6	- Optical smoke detectors (Gen. II, Type A)	571	1[a]	50 053 860	0	2.31E-11	3.14E-09	4.28E-08	1.00E-08	1.92E-08
1	- Optical smoke detectors (Gen. III, Type A)	9	1[a]	1 183 464	0	9.76E-10	1.33E-07	1.81E-06	4.23E-07	8.11E-07
5	- Optical smoke detectors (Gen. III, Type A)	88	1[a]	7 713 024	0	1.50E-10	2.04E-08	2.78E-07	6.49E-08	1.24E-07
3	- Optical smoke detectors (Gen. III, Type B)[b]	7600	1[a]	459 532 800	11	3.41E-09	1.94E-08	6.56E-08	2.50E-08	2.07E-08
3	- Optical smoke detectors (Gen. III, Type B)[c]	7595	1[a]	459 230 476	6	1.75E-09	1.05E-08	3.87E-08	1.42E-08	1.26E-08
1	- Rate-of-rise heat detectors (Gen. II, Type A)	95	1[a]	12 430 317	0	9.29E-11	1.27E-08	1.72E-07	4.03E-08	7.72E-08
5	- Rate-of-rise heat detectors (Gen. II, Type A)	42	1[a]	3 681 216	0	3.14E-10	4.27E-08	5.82E-07	1.36E-07	2.61E-07
3	- IR-flame detectors (Gen. III, Type A)	400	1[a]	24 240 000	1	3.15E-09	3.58E-08	2.09E-07	6.19E-08	7.81E-08
1	- Multi-criteria detectors (Gen. IV, Type A)	29	1[a]	3 787 085	0	3.05E-10	4.16E-08	5.66E-07	1.32E-07	2.53E-07
4	- Multi-criteria detectors (Gen. IV, Type A)	2424	1[a]	42 468 480	0	2.72E-11	3.71E-09	5.05E-08	1.18E-08	2.26E-08
5	- Multi-criteria detectors (Gen. IV, Type A)	335	1[a]	29 362 080	0	3.93E-11	5.36E-09	7.30E-08	1.71E-08	3.27E-08
6	- Multi-criteria detectors (Gen. IV, Type A)	0	1[a]	20 039 076	0	5.76E-11	7.85E-09	1.07E-07	2.50E-08	4.79E-08
1	- Multi-criteria detectors (Gen. IV, Type B)	121	1[a]	15 954 410	0	7.24E-11	9.86E-09	1.34E-07	3.14E-08	6.02E-08
5	- Multi-criteria detectors (Gen. IV, Type B)	130	1[a]	11 394 240	0	1.01E-10	1.38E-08	1.88E-07	4.40E-08	8.42E-08
3	- Aspirating smoke detectors (Gen. 1)[b]	60	1[a]	3 636 000	1	2.10E-08	2.39E-07	1.39E-06	4.13E-07	5.21E-07
6	- Aspirating smoke detectors (Gen. 1)[b]	91	1[a]	5 583 942	4	9.00E-08	5.77E-07	2.29E-06	8.06E-07	7.64E-07
5	- Aspirating smoke detectors (Gen. 2)[b]	6	1[a]	525 888	0	2.20E-09	2.99E-07	4.08E-06	9.52E-07	1.83E-06
6	- Aspirating smoke detectors (Gen. 2)[b]	93	1[a]	2 445 714	0	4.72E-10	6.43E-08	8.76E-07	2.05E-07	3.92E-07
1	- Manual call points (push buttons)	296	1[a]	49 362 686	0	2.34E-11	3.19E-09	4.34E-08	1.01E-08	1.94E-08
3	- Manual call points (push buttons)	430	1[a]	26 111 064	0	4.42E-11	6.03E-09	8.21E-08	1.92E-08	3.68E-08
4	- Manual call points (push buttons)	228	1[a]	37 189 608	0	3.11E-11	4.23E-09	5.76E-08	1.35E-08	2.58E-08
5	- Manual call points (push buttons)	2	1[a]	333 072	0	3.47E-09	4.72E-07	6.44E-06	1.50E-06	2.88E-06
6	- Manual call points (push buttons)	149	1[a]	12 579 210	0	9.18E-11	1.25E-08	1.70E-07	3.98E-08	7.63E-08

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

[b] Including all optical smoke detectors (Gen. III, Type B)

[c] Not including optical smoke detectors (Gen. III, Type B) inside restricted area.

However, up to the time being, the zero-failure statistics of the newer ASD does not lead to a lower mean failure rate, since the time observed for generation 2 ASD is much smaller than for those of generation 1. For manual alarm detectors no distinction in the detector generations was made since the basic principle of all types is the same over the whole observation period.

The unavailability of the fire detection system by self-signaling failures could not be quantified yet, but it is assumed to be quite small. Therefore, the ratio between unavailability from self-signaling failures and unavailability from inspection-related failures is not known. The lower the unavailability calculated based on the results from in-service inspection is, the more an in-depth analysis of the system's unavailability by self-signaling failures is needed.

Moreover, the applicability of the presented data for the reliability of fire detection systems is limited by the influence of the power supply. Fire detection systems are connected to the emergency power system and equipped with an additional battery, which makes the power supply reliable and redundant. However, one reportable event occurred in a German NPP, where the power supply adapter failed and the shift personnel did not recognize that the fire detection system switched to battery. As the current of the battery broke down, parts of the NPP area were without automatic fire detection until the failure was observed. Although the redundant power supply operated as designed during this event, the fire detection unavailability occurred under contribution of human factor which is not part of this study.

3.2 Fire dampers, smoke control equipment, and fire doors

The failure rates estimated for fire dampers, smoke control equipment and fire doors are listed in Table 3. Fire dampers are designed to close in case of fire. Most of them are connected to a ventilation duct; very few are installed in walls or ceilings as overflow opening between two rooms. Smoke control equipment refers to dampers and vents that are designed to open in case of fire. Dampers are installed in smoke extraction ducts or in inlet air ducts to extract smoke and increase air inlet flow. Vents are installed in roofs or walls leading directly to the outside without being attached to ducts.

Different types of fire dampers are installed in the plant units investigated. All fire dampers have got a thermal actuation mechanism, which is in the absolute most cases managed by a fusible link. Dampers in safety related plant areas can additionally be actuated remote controlled. The following remote controlled actuation types are present in the six plant units considered:

- 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 being actuated to close the dampers (open-circuit principle), additionally equipped with a pneumatic support to re-open the blade (type 2),
- lifting magnets which draw back a bolt when being 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).

The remote controlled actuation is redundant to the thermal actuation. Beside of the actuation, the blade has to move to the closed-position and must be structurally intact, which is evaluated by the 'closing/barrier function'. A fault tree for technical failures of fire dampers is shown in Figure 2. Since the remote controlled actuation mechanism is not present for all dampers, it is marked with a dashed line in Figure 2.

Table 3. **Failure rates of fire dampers, smoke control equipment and fire doors**

Plant	Active Fire Protection Systems and Components	Number of components	Test interval [years]	Observation period [h]	Number of failures	Failure rate [1/h] 5 % quantile	50 % quantile	95 % quantile	Mean value	Standard deviation
	Fire dampers [a]									
1	- Closing/Barrier function, type 0 + 3	795	1 [a]	132 336 303	16	1.76E-08	9.83E-08	3.22E-07	1.25E-07	1.00E-07
3	- Closing/Barrier function, type 0 + 2 + 3	1462	1	166 606 596	56	5.10E-08	2.75E-07	8.44E-07	3.39E-07	2.58E-07
4	- Closing/Barrier function, type 0 + 1 + 3	553	1/4	66 929 478	28	6.28E-08	3.41E-07	1.08E-06	4.26E-07	3.32E-07
5	- Closing/Barrier function, type 0 + 1 + 4	657	1 [a,b]	105 960 610	19	2.64E-08	1.46E-07	4.73E-07	1.84E-07	1.46E-07
6	- Closing/Barrier function, type 0 + 1 + 3 + 4	376	1/2	32 605 137	3	1.08E-08	7.45E-08	3.16E-07	1.07E-07	1.07E-07
	- Actuation:									
	- Remote controlled									
4	- Electro-pneumatic (type 1)	152	1/2	19 987 392	27	2.03E-07	1.10E-06	3.49E-06	1.38E-06	1.07E-06
5	- Electro-pneumatic (type 1)	322	1 [a,b]	53 554 472	2	3.83E-09	3.06E-08	1.44E-07	4.67E-08	5.06E-08
6	- Electro-pneumatic (type 1)	107	1/2	9 379 620	4	5.36E-08	3.44E-07	1.37E-06	4.80E-07	4.55E-07
3	- Lifting magnet + pneumatic reopening (type 2)	539	1	61 423 362	148	3.69E-07	1.98E-06	5.96E-06	2.42E-06	1.81E-06
1	- Lifting magnet (type 3)	641	1 [a]	106 764 960	9	1.18E-08	6.81E-08	2.36E-07	8.89E-08	7.53E-08
3	- Lifting magnet (type 3)	345	1	39 315 510	30	1.15E-07	6.23E-07	1.96E-06	7.75E-07	6.02E-07
4	- Lifting magnet (type 3)	221	1	30 996 576	28	1.36E-07	7.36E-07	2.33E-06	9.19E-07	7.16E-07
6	- Lifting magnet (type 3)	101	1/2	8 853 660	7	1.08E-07	6.35E-07	2.29E-06	8.47E-07	7.38E-07
5	- Magnetic clamp (type 4)	44	1 [a,b]	3 856 512	4	1.30E-07	8.36E-07	3.32E-06	1.17E-06	1.11E-06
6	- Magnetic clamp (type 4)	30	1/2	2 498 310	0	4.62E-10	6.30E-08	8.58E-07	2.00E-07	3.84E-07
1	- Thermal (soldered strut), type 0 + 3	724	1 [a]	95 203 104	0	1.21E-11	1.65E-09	2.25E-08	5.26E-09	1.01E-08
3	- Thermal (soldered strut), type 0 + 2 + 3	1462	1	100 738 872	98	1.47E-07	7.91E-07	2.40E-06	9.68E-07	7.29E-07
4	- Thermal (soldered strut), type 0 + 1 + 3	443	10	60 748 380	16	3.84E-08	2.14E-07	7.01E-07	2.72E-07	2.18E-07
5	- Thermal (soldered strut), type 0 + 1 + 4	465	1 [b]	40 756 320	7	2.35E-08	1.38E-07	4.96E-07	1.84E-07	1.60E-07
6	- Thermal (soldered strut), type 0 + 1 + 3 + 4	276	10	24 194 160	4	2.08E-08	1.33E-07	5.29E-07	1.86E-07	1.76E-07
	Smoke control equipment									
1	- Smoke extraction dampers in ducts	27	1	4 479 587	13	4.18E-07	2.35E-06	7.86E-06	3.01E-06	2.46E-06
3	- Smoke extraction dampers in ducts	98	1	17 166 192	36	3.16E-07	1.71E-06	5.33E-06	2.13E-06	1.64E-06
4	- Smoke extraction dampers in ducts	112	1	22 581 888	3	1.56E-08	1.08E-07	4.56E-07	1.55E-07	1.55E-07
5	- Smoke extraction dampers in ducts	59	1	5 171 232	1	1.47E-08	1.68E-07	9.79E-07	2.90E-07	3.66E-07
6	- Smoke extraction dampers in ducts	28	1/2	2 454 144	0	4.71E-10	6.41E-08	8.73E-07	2.04E-07	3.91E-07
1	- Smoke extraction vents (roof-installed)	11	1	1 446 456	2	1.42E-07	1.13E-06	5.33E-06	1.73E-06	1.87E-06
4	- Smoke extraction vents (roof-installed)	35	1	6 969 144	18	3.79E-07	2.10E-06	6.82E-06	2.65E-06	2.12E-06
1	- Smoke extraction vents (wall-installed)	6	1	788 976	1	9.66E-08	1.10E-06	6.42E-06	1.90E-06	2.40E-06
5	- Smoke extraction vents (wall-installed)	4	1	350 592	0	3.30E-09	4.49E-07	6.11E-06	1.43E-06	2.74E-06
	Fire doors [a]									
4	- Barrier function	527	1	68 879 538	1	1.11E-09	1.26E-08	7.35E-08	2.18E-08	2.75E-08
6	- Barrier function	387	1/2	33 924 420	2	6.05E-09	4.82E-08	2.27E-07	7.37E-08	7.99E-08
4	- Self-closing function	596	1	77 950 692	105	2.05E-07	1.11E-06	3.35E-06	1.35E-06	1.02E-06
6	- Self-closing function	459	1/2	40 235 940	29	1.08E-07	5.88E-07	1.85E-06	7.33E-07	5.70E-07
4	- Self latching function	596	1	77 950 692	93	1.81E-07	9.79E-07	2.97E-06	1.20E-06	9.04E-07
6	- Self latching function	459	1/2	40 235 940	20	7.33E-08	4.05E-07	1.31E-06	5.09E-07	4.04E-07
4	- Door-coordinator function	69	1	9 071 154	15	2.40E-07	1.34E-06	4.43E-06	1.71E-06	1.38E-06
6	- Door-coordinator function	72	1/2	6 311 520	14	3.20E-07	1.80E-06	5.98E-06	2.30E-06	1.87E-06
1	- Release by hold-open device	138	1/4	22 903 743	5	2.84E-08	1.75E-07	6.67E-07	2.40E-07	2.19E-07
3	- Release by hold-open device	181	1/4	32 892 264	7	2.91E-08	1.71E-07	6.15E-07	2.28E-07	1.99E-07
4	- Release by hold-open device	68	1/4 [c]	13 649 064	9	9.24E-08	5.32E-07	1.85E-06	6.96E-07	5.89E-07
5	- Release by hold-open device	22	1/4 [c]	3 584 904	5	1.82E-07	1.12E-06	4.26E-06	1.53E-06	1.40E-06
6	- Release by hold-open device	28	1/4 [c]	2 150 837	5	3.03E-07	1.87E-06	7.11E-06	2.56E-06	2.34E-06

[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 is 1 year, sometimes 3 months.

[c] The most common test interval for hold-open devices is 3 months, sometimes with additional short tests every month.

All dampers can be manually operated from only one side of the fire barrier by means of a test button. However, this actuation mechanism is not accessible in many cases during plant operation and is therefore marked with a dotted line in Figure 2. A failure of the fire damper occurs in case of failure of either all actuation mechanisms being present or in case of failure of the closing/barrier function.

The most common test interval for fire dampers is one year, sometimes it is six months. In some plants functional testing of the manual and remote controlled actuation is separated such that in average the damper blade is moved every six months. 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 issued as a result of findings at fire dampers in the mid-1990es. Meanwhile, in some plants these inspections are carried out periodically every ten years by a destructive inspection, where the fusible link is molten by a hot air dryer.

In other plants, the fusible link is removed from the damper within the yearly inspections in order to simulate a molten fusible link and to test the mechanical closings components without destroying the fusible link.

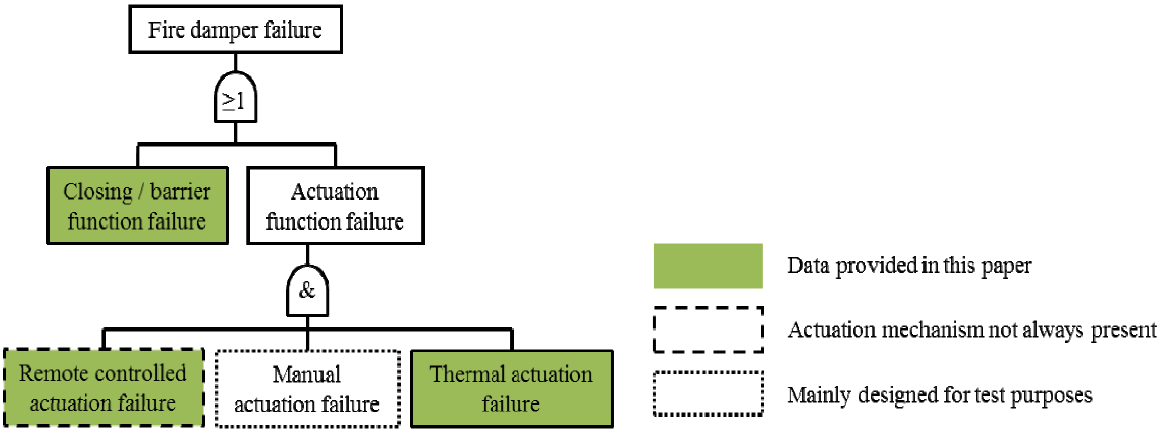


Figure 2. **Fault tree for technical failures of fire dampers [11]**

Concerning the closing/barrier function, in total 117 failures have been observed in the six power plant units to be investigated. Separating this number to account for plant specific data the findings show that each plant has about the same number of failures taking into account the different observation periods and number of components. The majority of these failures occurred because of dust deposit or resinified oil on the mobile inner parts of the dampers which blocked the closing function. A small number of failures were caused by significant damages of the damper blades.

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 place) to the damper is covered. Typical failure modes were 'jammed', 'stiff", or 'did not close'. Failures are always assigned to the dampers, even if the failure is located at the trigger, because the trigger is not modelled. The mean failure rates of the different types of dampers ranges from 10^{-6} to 10^{-8} h^{-1}. Although the actuation mechanism types 2 and 3 are based on the outdated open-circuit principle, the reason for the differences in the failure rate is rather plant specific, since the same actuation type shows the same range (10^{-6} to 10^{-8} h^{-1}) as mentioned above.

Moreover, the functional unavailability of thermal actuation by a fusible link is in the same order of magnitude in the different plants (except for plant 1 which had no recorded failures during the observation period). The mean value of the failure rate for plant 1 is λ = 5.26 E-09 h-1, which is the lowest value of all mean failure rates for the different types of actuation in the entire plants. However, as the test interval is about ten times longer than that for the remote controlled actuation, the resulting unavailability per demand may increase up to the upper boundary in comparison to all other actuation mechanisms.

Concerning the reliability of smoke extraction equipment no fault tree has been developed, since the majority of such equipment does not have redundant technical actuations. Also, the additional possibility of a manual opening by the fire brigade under operational conditions cannot be statistically evaluated.

Regarding smoke extraction dampers in ducts, which are mostly modified fire dampers, again the observed failure types were similar to those of fire dampers. A failure of a smoke extraction damper was assumed if it did not open. In case it opened but did not latch in open position, this observation was only interpreted as deficiency.

Smoke extraction vents which are not attached to ducts are also common in non-nuclear buildings. It was distinguished between the installation of vents in roofs and in walls. Failure types of the vents were empty CO_2-cartridges or leakages at the pneumatic pipework. A stiff frame and damaged wire ropes also occurred and were accounted for as failures. The different plant units show a variety of failures of smoke extraction dampers and vents. The mean values of the plant specific reliability data for these components are within one order of magnitude.

A notable limit of application of the data for all remote controlled fire dampers as well as smoke extraction dampers concerns the already mentioned fact that the observed failures were not assigned to the triggers, but only to the dampers themselves. That implies that so far realistic data for more than one damper of a fire compartment not operating as designed cannot be provided yet.

For fire doors a fault tree (see Figure 3) was developed to consider the different configurations such as single and double winged doors with additional door-coordinator, and doors equipped with or without hold-open device. The fault tree is similar to the one for the fire dampers (cf. Figure 2), however the barrier and the closing function are separated and closing can be achieved manually or automatically. For manual closing no data have been provided in this study. The automatic closing depends on the availability of the door closer(s), for double winged doors of the door-coordinator and, if present, of the availability of the automatic release function of the hold-open device.

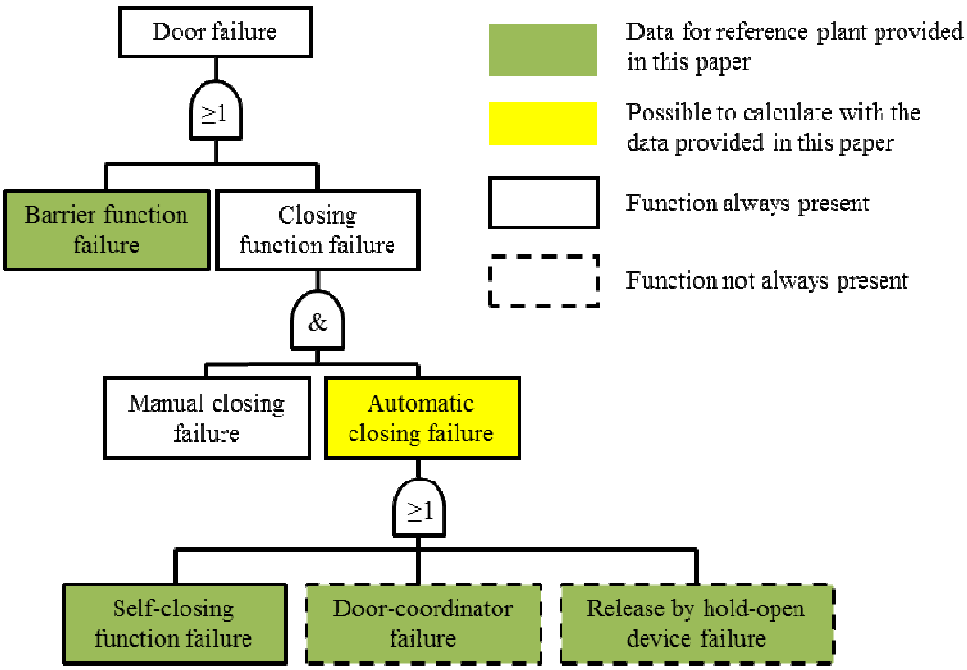


Figure 3. **Fault tree for technical failures of fire doors**

Slightly better values (both number of failures and mean failure rate) were gained for plant 6 in comparison to plant 4 (except the barrier function) as presented in Table 2. Furthermore, Table 2 demonstrates that the number of malfunctions of the hold-open devices was nearly the same in both plants, resulting in fairly similar mean failure rates. Barrier failures were accounted for, if an obstacle not easy to remove was found in the door or when all three sealing strips of the door were missing. The number of failures is correlated with the number of doors with no distinction between single and double winged doors. The self-closing function is related to the number of door wings, because each wing has its own door closer. The self-closing function was extended by an additional possible failure of the self-latching function, because many door wings close properly by moving to the closed position without latching. It depends on door direction and fire dynamics, whether pressure increase in the fire compartment may open the fire door. The door-coordinator function is only correlated to double wing doors. The analyst applying the data may decide whether one or both wings of a double wing door are opened. Finally, the release function of hold-open devices is analyzed. Typical failures of the hold-open device resulted from malfunctions of the fire detectors. The barrier function of the door is highly reliable compared to the closing function (see Figure 3) depending on different sub-functions. A large uncertainty with respect to the door failure is due to the fact that for manual closing no data have been provided. However, considering the technical reliability data provided in this paper a worst case scenario can be calculated, disregarding the likely event of human interaction (manually closing door).

3.3 Fire extinguishing systems and equipment

The water based fire extinguishing systems and equipment (water deluge systems and hydrants) in the plants under investigation have got a fire water main ring for the water supply. The reliability of the main ring, including, if present, flooding valves as well as isolation valves of buildings were out of scope of the investigations and have to be considered separately. To achieve the necessary water supply for the deluge systems and hydrants, the main ring of a plant is connected with a number, of water pumps, which varies from plant to plant. The plant specific data for fire water pumps, water deluge systems, and hydrants of different types are presented in Table 4. The following failure types of the fire water pumps have been observed:

- for the remote controlled actuation of the pumps the safety hatch of the actuation drawer standing in an intermediate position, when being interrupted on command, and
- for the operation of the pumps insufficient water volume or water pressure.

During the observation period, four findings were revealed at the water pumps not reaching a sufficient water pressure/flux and two findings, where the remote actuation from the main control room failed. More specifically, in plant 2 findings have been observed and coded as failures of the pump itself, due to insufficient water pressure/flow. One further finding interpreted as failure was the failure of the remote controlled actuation from the main control room. In plant 5, another two failures of the pump itself occurred, causing too low water pressure/volume in the system. The only other failure of the required function was a failure of the remote control actuation of a pump in plant 6.

The nuclear power plants, for which the investigations were performed, rely on fixed automatically and/or manually actuated water deluge systems (in the following called deluge systems) for fire extinguishing in areas important to safety. The deluge systems are equipped with the following types of valve stations actuated by remote control:

1.1 hydraulically actuated butterfly valves controlled (open/close) via a magnetic (4/2-way) valve (installed in plant 4),

1.2 hydraulically actuated butterfly valves controlled (open/close) via a magnetic (4/2-way) valve; in addition manually operable by means of a square socket key (installed in plants 1 and 3),

2. hydraulically operated poppet valves controlled (open/close) via a magnetic valve (in place in plant 6),
3. electromagnetically controlled (only open) valves with manual override (in place in plant 6),
4. electric motor operated valves with manual override (installed in plant 5), and
5. pneumatically preserved valves, which are controlled (only open) by discharge of the trigger line (in place in plants 1 and 5).

The most common test interval for each type is six months; however longer and shorter test intervals have also been used during the observation period (see also Table 4).

Findings that have been interpreted as failures of the required function of the valve stations actuated via remote control are e.g.:

- "*does not open or opens first after 2nd or 3rd trial*",
- "*magnetic valve opens first after cleaning the relief pipe*",
- "*double filter blocked*",
- "*turn actuator stuck*",
- "*4/2-way valve stuck, no pressure to trigger*",
- "*actuation valve did not actuate after manual and remote controlled actuation*".

Table 4. **Failure rates for fire extinguishing systems and equipment**

Plant	Active Fire Protection Systems and Components	Number of components	Test interval [years]	Observation period [h]	Number of failures	Failure rate [1/h] 5 % quantile	50 % quantile	95 % quantile	Mean value	Standard deviation
	Water pumps									
1	- Pump (incl. motor)	4	1/2	666 240	0	1.73E-09	2.36E-07	3.22E-06	7.52E-07	1.44E-06
3	- Pump (incl. motor)	4	1/2[e]	698 400	0	1.65E-09	2.25E-07	3.07E-06	7.17E-07	1.37E-06
4	- Pump (incl. motor)	6	1/4[f]	1 209 744	2	1.70E-07	1.35E-06	6.37E-06	2.07E-06	2.24E-06
5	- Pump (incl. motor)	2	1/2[g]	333 072	2	6.16E-07	4.91E-06	2.31E-05	7.50E-06	8.14E-06
6	- Pump (incl. motor)	5	1/4	438 300	0	2.64E-09	3.59E-07	4.89E-06	1.14E-06	2.19E-06
1	- Remote actuation	4	1/2	666 240	0	1.73E-09	2.36E-07	3.22E-06	7.52E-07	1.44E-06
3	- Remote actuation	4	1/2[e]	698 400	0	1.65E-09	2.25E-07	3.07E-06	7.17E-07	1.37E-06
4	- Remote actuation	6	1/4[f]	1 209 744	1	6.30E-08	7.18E-07	4.19E-06	1.24E-06	1.57E-06
5	- Remote actuation	2	1/2[g]	333 072	0	3.47E-09	4.72E-07	6.44E-06	1.50E-06	2.88E-06
6	- Remote actuation	5	1/4	438 300	1	1.74E-07	1.98E-06	1.16E-05	3.42E-06	4.32E-06
	Water deluge systems									
4	- Remote controlled valve stations, type 1.1	58	1/4[a]	11 694 192	70	9.10E-07	4.91E-06	1.50E-05	6.03E-06	4.57E-06
1	- Remote controlled valve stations, type 1.2	26	1/2	4 330 560	2	4.74E-08	3.78E-07	1.78E-06	5.77E-07	6.26E-07
3	- Remote controlled valve stations, type 1.2	108	1/2	18 927 672	49	3.92E-07	2.12E-06	6.52E-06	2.61E-06	2.00E-06
6	- Remote controlled valve stations, type 2	3	1/2[b]	262 980	0	4.39E-09	5.98E-07	8.15E-06	1.90E-06	3.65E-06
6	- Remote controlled valve stations, type 3	30	1/4	2 636 484	2	7.78E-08	6.21E-07	2.92E-06	9.48E-07	1.03E-06
5	- Remote controlled valve stations, type 4	21	1/2[c]	3 497 256	4	1.44E-07	9.22E-07	3.66E-06	1.29E-06	1.22E-06
1	- Remote controlled valve stations, type 5	5	5	657 480	0	1.76E-09	2.39E-07	3.26E-06	7.62E-07	1.46E-06
5	- Remote controlled valve stations, type 5	4	1/2[d]	666 144	0	1.73E-09	2.36E-07	3.22E-06	7.52E-07	1.44E-06
	Hydrants									
1	- Field hydrants	38	1a	6364345	2	3.22E-08	2.57E-07	1.21E-06	3.93E-07	4.26E-07
3	- Field hydrants	49	6 m	8 484 528	1	8.99E-09	1.02E-07	5.97E-07	1.77E-07	2.23E-07
4	- Field hydrants	41	1a	8 424 360	0	1.43E-10	1.95E-08	2.66E-07	6.20E-08	1.19E-07
5	- Field hydrants	17	1a	2 208 768	2	9.28E-08	7.41E-07	3.49E-06	1.13E-06	1.23E-06
6	- Field hydrants	13	1a	911 664	0	1.27E-09	1.73E-07	2.35E-06	5.49E-07	1.05E-06
1	- Wall hydrants	140	1a	23 379 764	0	4.94E-11	6.73E-09	9.17E-08	2.14E-08	4.11E-08
3	- Wall hydrants	205	1a	35 745 480	0	3.23E-11	4.40E-09	6.00E-08	1.40E-08	2.69E-08
4	- Wall hydrants	146	1a	29 437 104	0	3.92E-11	5.35E-09	7.28E-08	1.70E-08	3.26E-08
5	- Wall hydrants	132	1a	17 312 285	1	4.40E-09	5.02E-08	2.92E-07	8.67E-08	1.09E-07
6	- Wall hydrants	86	1a[h]	6 031 008	0	1.92E-10	2.61E-08	3.55E-07	8.30E-08	1.59E-07
3	- Foam wall hydrants, foam mixing function	26	1a	2 944 032	3	1.20E-07	8.25E-07	3.50E-06	1.19E-06	1.19E-06
4	- Foam wall hydrants, foam mixing function	9	1a	1 262 304	4	3.98E-07	2.55E-06	1.02E-05	3.56E-06	3.38E-06
5	- Foam wall hydrants, foam mixing function	28	1a	2 454 144	3	1.44E-07	9.90E-07	4.19E-06	1.43E-06	1.42E-06

[a] The most common test interval for water deluge systems, type 1.1 is 3 months, sometimes every 6 weeks or every year.

[b] The most common test interval for water deluge systems, type 2 is 6 months, sometimes every 3 months.

[c] The most common test interval for water deluge systems, type 4 is 6 months, sometimes every 3 months or every year.

[d] The most common test interval for water deluge systems, type 5 is 6 months, sometimes every week or every 1 or 2 years.

[e] The most common test interval for water pumps in plant 3 is 6 months, sometimes once a month.

[f] The most common test interval for water pumps in plant 4 is 3 months, sometimes once a month.

[g] The most common test interval for water pumps in plant 5 is 6 months, sometimes once a week or every 1 year.

[h] The most common test interval for wall hydrants in plant 6 is 1 year, sometimes 3 months.

In this context, it has to be mentioned that for some of the plants under consideration the manual actuation of the deluge system is not explicitly tested according to the fact that in such a test a by-pass valve which cannot be closed for test purposes by another valve would be opened. The only possibility for testing the manual actuation is to close the main pipe valve supplying several valve stations of the deluge system. This possibly might affect the fire safety concept for several redundant trains at the same time. Furthermore, the reliability of the deluge systems´ water pipes and nozzles has not been regarded in the study.

Findings such as delayed opening (few seconds), leakages or findings concerning the closing function after opening of the main valve were assessed as deficiencies. The mean values, standard deviations and quantiles for the different types of the deluge system valve stations being actuated via remote control are presented in Table 4.

There are two different types of hydrants installed in the plant; field and wall hydrants. A number of the wall hydrants are additionally equipped in order to add foam to the water. Five findings of the in-service inspections were interpreted as functional failures of field hydrants and one as a failure of a wall hydrant. The failures were e.g. stuck drop jacket or stuck main valve. The foam mixing function was assessed individually resulting in a total of ten findings that could be interpreted as failures. A typical failure of the foam mixing function was a stuck foam mixing valve.

Stiffness of valves concerning all hydrants has been interpreted as deficiency only, because such problems are normally easy to compensate by the tools of the on-site fire brigade.

4. Conclusions

The technical reliability data for active fire protection systems and equipment established in the past for six NPP units of different type and age have recently been updated and further extended. The data have been evaluated plant specifically, mainly by analyzing results of periodic in-service inspections covering approx. 111 years of plant operation. In addition, updates of the generic data have been provided (cf. [7] and [14]). The updated generic reliability database will be included in an additional document supplementing the existing technical document on PSA Data [3] to be issued in 2014 and can be also applied in the frame of Fire PSA for NPP outside Germany.

According to the fact that the data is based on results of in-service inspections, the given failure rates do not cover design failures or unavailability of components that occurred and were repaired in-between two inspections. Design failures may concern the right selection of fire detectors, a sufficient rating of fire barrier elements, or the selection of suitable fire extinguishing agents.

Acknowledgements

The extended update of the existing technical reliability data for active fire protection features has been funded by the Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit, BMUB) within the frame of the project 3610R01370. Furthermore, the authors are grateful for the support of the personnel of the German nuclear power plants ‘Gemeinschaftskernkraftwerk Grohnde, Emmerthal’, ‘Gemeinschaftskraftwerk Neckar, Neckarwestheim’, ‘Kernkraftwerk Brunsbüttel, Brunsbüttel’, ‘Kernkraftwerk Gundremmingen, Gundremmingen’, and ‘Kernkraftwerk Philippsburg, Philppsburg’, involved in the project, for providing data and additional information. Finally, the authors want to thank Jörg Peschke (GRS) for fruitful discussions in generating failure rate distributions.

References

[1] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) (2013), Safety Requirements for Nuclear Power Plants, Federal Gazette, January 24, 2013 (in German), http://www.bmu.de/service/publikationen/downloads/details/artikel/bekanntmachung-der-sicherheitsanforderungen-an-kernkraftwerke-vom-22-november-2012/?tx_ttnews%5BbackPid%5D=266

[2] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke (2005), Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: August 2005, BfS-SCHR-37/05, Salzgitter, Germany, October 2005

[3] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke (2005), Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: August 2005, BfS-SCHR-38/05, Salzgitter, Germany, October 2005

[4] Röwekamp, M., T. Riekert, W. Sehrbrock Ermittlung von Zuverlässigkeitskenngrößen für Brandschutzeinrichtungen in deutschen Kernkraftwerken, Schriftenreihe Reaktorsicherheit und Strahlenschutz, BMU-1997-486, ISSN 0724-3316, Bonn, Germany, March 1997

[5] Röwekamp, M., S. Oltmanns (2001), Ermittlung kernkraftwerksspezifischer Zuverlässigkeitskenngrößen für Brandschutzeinrichtungen in einem älteren Kernkraftwerk und in einer Konvoi-Anlage, Schriftenreihe Reaktorsicherheit und Strahlenschutz, BMU-2001-573, ISSN 0724-3316, Bonn, Germany, 2001

[6] von Linden, J., et al. (2005), Ausgewählte probabilistische Brandanalysen für den Leistungs- und Nichtleistungsbetrieb einer Referenzanlage mit Siedewasserreaktor älterer Bauart, Schriftenreihe Reaktorsicherheit und Strahlenschutz, BMU-2005-666, Bonn, Germany, http://www.bmu.de/files/strahlenschutz/schriftenreihe_reaktorsicherheit_strahlenschutz/application/pdf/schriftenreihe_rs666.pdf, 2005

[7] Forell, B., and S. Einarsson (2014), Ergänzung und Aktualisierung von Zuverlässigkeitskenngrößen für Brandschutzeinrichtungen in deutschen Leichtwasserreaktoren, GRS-A-3719, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany, under publication 2014

[8] Peschke, J. (1997), Der Superpopulationsansatz zur Ermittlung von Verteilungen für Ausfallraten und Eintrittshäufigkeiten auslösender Ereignisse, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-A-2444, Garching, Germany, April 1997

[9] Peschke, J. (2010), Methodik zur Berücksichtigung epistemischer Unsicherheitsquellen bei der Schätzung von Zuverlässigkeitskenngrößen, Technical Report, GRS-A-3540, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Garching, Germany, April 2010

[10] Stiller, J. C., A. Kreuser, and C. Verstegen (2008), Consideration of Additional Uncertainties in the Coupling Model for the Estimation of Unavailabilities due to Common Cause Failures, in: Proceedings of the 9th International Conference on Probabilistic Safety Assessment & Management PSAM 9, Hong Kong, China, May 2008

[11] Forell, B., S. Einarsson, M. Roewekamp, and H.-P. Berg (2012), Updated Technical Reliability Data for Fire Protection Systems and Components at a German Nuclear Power Plant, in: 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012 (PSAM11 ESREL 2012), ISBN: 978-1-62276-436-5, Curran Associates, Inc., Red Hook, NY, USA, 2012, pp. 3783-3794

[12] Comité Européen des Assurances (CEA) (1999), Specifications for Fire Detection and Fire Alarm Systems Requirements and Test Methods for Aspirating Smoke Detectors, CEA 0422, Brussels, Belgium, 1999

[13] European Standard (2006), EN 54-20: Fire detection and fire alarm systems – Part 20: Aspirating smoke detectors, 2006

[14] Forell, B., S. Einarsson, and M. Röwekamp (2014), Technical Reliability of Active Fire Protection Features – Generic Database Derived from German Nuclear Power Plants, Paper No. 230, in: Proceedings 12th International Probabilistic Safety Assessment and Management Conference, Honolulu, HI, USA, in preparation, 2014

Insights and Opportunities from the EPRI Updated Fire Events Database

Ashley Lindeman
Electric Power Research Institute, United States of America

Abstract

The EPRI updated Fire Events Database (FEDB) was published in July 2013. This update adds a decade of fire event history for nuclear power plant facilities to the previous database. The revised FEDB structure is improved by providing expanded database fields, a more comprehensive incident detail and better traceability for data sources. The intent of this paper is to discuss the structure and organization of the database, describe future update plans and present insights from the most recent decade of data. Previously, the FEDB was used primarily for generation of fire ignition frequencies and non-suppression probabilities to support fire probabilistic risk assessments (PRAs). While the new FEDB will allow these parameters to be updated, the more detailed event data assembled in the FEDB will be used to inform other aspects of fire modelling and treatment in PRAs

1. Introduction

In 2010 EPRI undertook a major effort to update the fire events database (FEDB). The previous update of the fire events database included fire events up to the year 1999. The oldest records in the fire events database are from 1968. The goal of the project was to update and improve the database structure and content as well as to capture the most recent decade of operating experience. This updated database, which was published in July 2013, serves as the primary repository for fire incident data for US commercial nuclear power plants.

1.1 The EPRI updated fire events database

A report summarizing the updated FEDB was published by EPRI in July 2013 as EPRI 1025284 [1]. The actual database remains proprietary to EPRI. The database includes fire event data from Nuclear Electric Insurance Limited (NEIL), licensee event reports (LERs), event notifications and plant incident fire reports. A challenge to this project was that EPRI needed to collect a significant amount of incident data, reflecting experience over 10 years, from the plants. This effort required the assistance of plant staff to sort through event reports and provide summary descriptions on keyword searches, including smoke, fire, smolder, burn, explosion and extinguish. This labor-intensive process was implemented to help ensure that no events that could have been fires were overlooked. This initial search yielded approximately 1000 to 5000 event records representing possible fires for each plant. Due to the large amount of data collected, the help of the Boiling Water Reactor Owner's Group (BWROG) and Pressurized Water Reactor Owner's Group (PWROG) was enlisted to screen the search results to identify actual fires. After this review, approximately 100 to 300 potential fire events remained for each plant. At the conclusion of this screening, each plant was contacted to provide full reports on the remaining events. EPRI reviewed these reports, and further evaluation led to the inclusion of approximately 5 to 50 fires per plant in the FEDB.

Under a memorandum of understanding (MOU), the US Nuclear Regulatory Commission Office of Research and their contractors were invited to participate in several audits. The purpose of these audits was to assure accuracy and completeness in the FEDB, and to reach concurrence on fire severity classification. Three audits were conducted over the course of the project, each focusing on critical aspects of the database, including: data acquisition and screening, process modifications and initial implementation of fire severity classification, and final severity classification and implementation of the override rules [1].

In addition to providing more recent data to update fire frequencies and non-suppression probability estimates, the FEDB was intended to:

- Improve data quality for FPRA applications
- Establish a system to collect data and update fire ignition frequencies on a continual or periodic basis
- Provide data to support improvements in the modeling of non-suppression probabilities
- Provide data to support enhanced fire ignition frequency modeling
- Provide improved fire event severity characterization and classification to allow uncertainty in estimates of damaging fire frequencies to be reduced
- Provide estimates of key occurrence rates discriminated by fire severity
- Provide fire incident benchmarks to help test, adjust and validate fire growth and severity / damage modeling assumptions.

1.2 Future fire events database updates

Realizing that the collection of fire event incident data is an ongoing endeavour, EPRI has worked with the Institute for Nuclear Power Operations (INPO) to collect data on an ongoing basis, rather than relying on less efficient and labor-intensive retroactive processes. INPO collects fire incident data in its INPO Consolidated Events System (ICES) database. INPO requested plants submit fire incidents for the 2010-2013 time period in which the EPRI FEDB scope ended. Currently, the fire event reporting is active, and plants submit fire reports as incidents occur. This reporting is advantageous as it eliminates duplicate reporting and the need to collect this data long after the events occurred, when recollection of what transpired may not be as precise. Additionally, this method allows INPO to follow up with any plant that may have conflicting information or missing fields in the initial fire report. While INPO may perform its own trending, counting and analysis, EPRI will receive the raw data to make the fire severity classification and to incorporate it into the fire PRA data.

EPRI will receive the fire incident data collected in the ICES database annually. EPRI will update the fire ignition frequencies and non-suppression probabilities at three- to five-year intervals.

2. Current Fire PRA Application

Once events are fully imported into the fire events database, fire severity is initially classified using an algorithm. This algorithm, described in EPRI 1025284 [1], classifies fire events as challenging, potentially challenging, or non-challenging using well defined criteria. Override criteria are provided for specific situations where, for example, a non-challenging classification would be more appropriate than potentially challenging. An example of the application of the override criteria would be to eliminate fires in lighting ballasts and in wall outlets and switches that are not credible sources of fires that could affect equipment important to safety in a nuclear power plant. Trained analysts review the preliminary fire severity classification from the algorithm. Final fire severity is determined once the analyst and other reviews (including by the NRC) are complete. Any differences of opinion between the EPRI classification and NRC classification are noted.

Potentially challenging (PC) and challenging (C) fires move on to the next step of determining fire ignition frequencies and non-suppression probabilities (NSPs). NUREG/CR-6850 [2] divides potential plant ignition sources into one of 37 generic fire ignition frequency bins. These bins represent fires originating from both fixed (e.g., diesel generators and pumps) and transient sources. Each PC and C fire is reviewed to determine the most appropriate fire ignition frequency bin. The method for determining the binning of events relative to non-suppression follows a similar process. Non-suppression events are binned into different suppression probability curves based on location and/or type of fire. The potential binning options include transients, welding, electrical, cable, oil, flammable gas, transformer yard, containment, control room, turbine generator and high energy arcing faults. There are some unique rules, such that all fires that occur in the control room are included in the main control room (MCR) bin regardless of their source because these areas are different from other plant areas (i.e., the MCR is continuously occupied).

Currently, the fire ignition frequencies and non-suppression probabilities are being revised to include fire events through the year 2009. This project is being led by EPRI and the NRC is involved in the review of frequency and non-suppression events. A joint EPRI-NRC report with the new data will be available in mid-2014.

3. Exercising the Fire Events Database

Exploration of the database has revealed that a large percentage of the fires counted in fire ignition frequencies do not align with the heat release profiles prescribed in Table G-1 [2]. Most of the fires in the database are generally smaller than the postulated heat release rates. This is especially true of electrical cabinet fires. Further research is needed to connect the size of fires counted in the fire frequencies with the heat release rates postulated using the current NUREG/CR-6850 method. In the near term, EPRI and the NRC will collaborate and revise heat release rates and improve modeling techniques for estimating damage due to electrical cabinet fires. The updated FEDB, along with the recently completed NRC test series, will be a critical input to inform this project.

3.1 Binning of electrical cabinets

Electrical cabinets within nuclear power plants come in a variety of sizes, functions and combustible loading. In terms of ignition frequency, plant cabinets of sufficient size [2], are binned into one of the following categories:

- Main Control Boards (Bin 4)
- Battery Chargers (Bin 10)
- Electrical Cabinets (Bin 15)
- Junction Boxes (Bin 18)

While main control boards, battery chargers, and junction boxes are well defined items, most of the electrical cabinets fall into bin 15. This bin contains electrical equipment such as switchgear, motor control centers, DC distribution panels, relay cabinets, control panels, fire protection panels, and any electrical cabinets not otherwise assigned to bins 4, 10 or 18.

Current PRA results indicate that electrical cabinets can be a dominant contributor to plant risk. While it is expected that sources such as the main control board would contribute to plant risk, a high number of electrical cabinet (bin 15) fires are often found in the top fire scenarios. There is an increased need to understand, from an ignition frequency and fire dynamics standpoint, the treatment of these fires. With more specific coding fields in the Updated FEDB as well as better traceability to references such as LERs, event notifications, and plant reports, a better understanding and discrimination of the events can be achieved.

A refinement to Table G-1 of NUREG/CR-6850 [2] was published as EPRI 1022993 [3]. This refinement differentiated electrical cabinets by fuel density. The fuel loading of the cabinets was found to be an important determinant of peak heat release rate. EPRI 1022993 [3] suggests differentiating cabinets into either a low volumetric or high volumetric fuel load. Table 1, extracted from Reference 3, provides guidance based on cabinet type to determine classification. Using the Updated FEDB, bin 15 cabinets events (from 1990-2009) are explored in more detail to determine if the ignition source could be binned with either a low or high volumetric density factor. Further review addressed whether electrical cabinets that experienced fires could be of power, control or instrumentation function. The last exercise explored the specific type of cabinet (i.e. motor control center or switchgear) to see if any outliers exist.

Table 1. **Determination of Cabinet Fuel Load per EPRI 1022993**

Cabinets with Low Volumetric Fuel Density Factor	**Cabinets with High Volumetric Fuel Density Factor**
Switchgear, motor control centers, load centers, inverters, battery chargers, distribution panels, and control panels containing a marginal amount of connectors, small gauge wiring and electronics	Electronic cabinets, relay racks, annunciator panels, main control boards, solid state cabinets and control panels containing a significant amount of connectors, small gauge wiring and electronics

The review of cabinets for volumetric fuel density type was generally straightforward. Most event reports provided sufficient detail to allow classification by equipment type. A few assumptions were made to determine volumetric fuel density classification. To be conservative, all control cabinets were classified as having a high volumetric fuel density factor. Typically, this information could not be determined without physically opening and examining the cabinet in question (which could present a challenge to plant safety). The second assumption was that all wall mounted cabinets had a low volumetric fuel density factor, with the justification that these cabinets are generally small and not likely to contain a high density of combustibles. The experience base was divided into two decades for clarity. The results of this review indicate that most fires have occurred in cabinets with a low volumetric fuel density as reported in Table 2.

Table 2**. Examination of Countable Electrical Cabinet Fires to Determine High / Low Volumetric Fuel Density (1990-2009)**

	1990 – 1999	**2000 - 2009**	**Total**
High Volumetric Fuel Density Factor	4	9	13
Low Volumetric Fuel Density Factor	32	20	52
Unknown	2	1	3

A second review was conducted to determine if new insights can be drawn from differentiating cabinet type by high level function. The high level function would be power (MCCs, switchgear, load centers, distribution cabinets, etc), control (relay racks, control cabinets, etc), and instrumentation (generally smaller, low-voltage cabinets). This review found that most fires originated in power cabinets. This is consistent with the first review as power cabinets are generally binned in the low volumetric fuel density factor. Although a few cabinet functions could not be determined, it was interesting to note that there were no countable fires in instrumentation cabinets. The results of this review are reported in Table 3.

Table 3. **Examination of Countable Electrical Cabinet Fires to Determine Cabinet Function (1990-2009)**

	1990-1999	**2000-2009**	**Total**
Power	31	19	50
Control	5	10	15
Instrumentation	0	0	0
Unknown	2	1	3

A further review of the ignition frequency data resulted in binning by specific cabinet type. Fortunately, most of the supporting information was sufficient to determine specific cabinet type. The results of this review are noted in Table 4 and graphically in Figure 1.

Table 4. **Examination of Countable Electrical Cabinet Fires to Determine Specific Cabinet Type (1990-2009)**

	1990-1999	**2000-2009**	**Total**
Motor Control Center	21	5	26
Switchgear / Load Center	6	6	12
Distribution panel or other power cabinet	4	6	10
Control Cabinet	3	6	9
Relay Rack	1	2	3
Inverter	0	2	2
Wall Mounted Cabinet	1	1	2
Unknown	2	1	3
Other	0	1	1

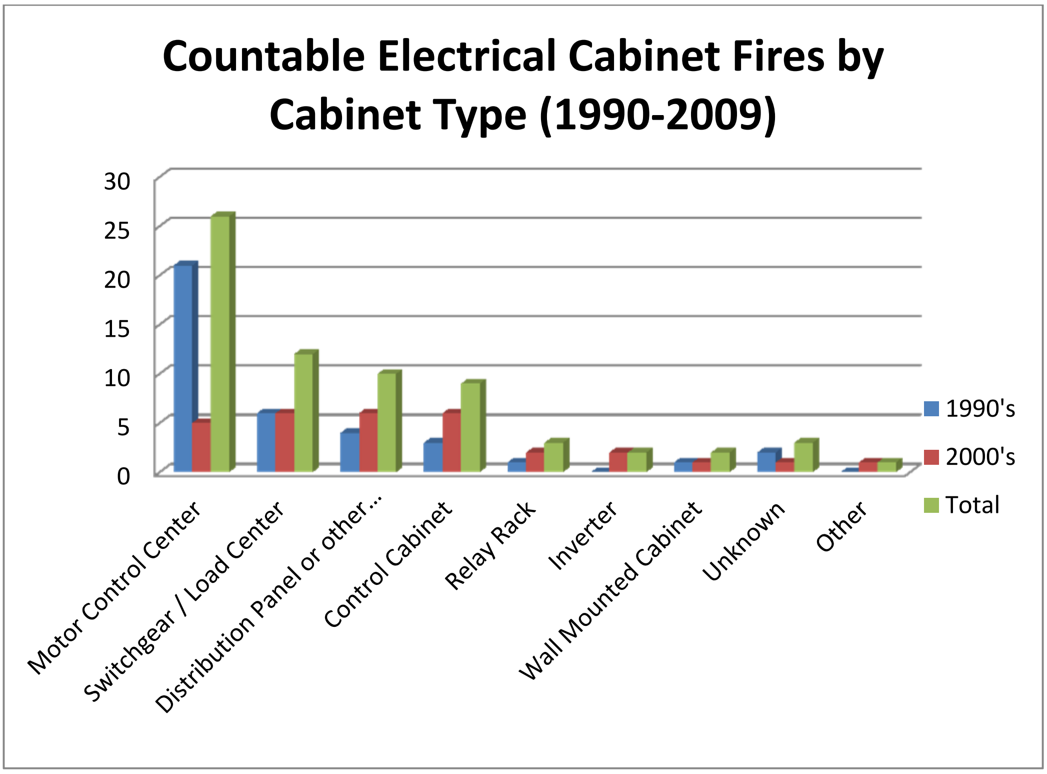


Figure 1. **Graph of Electrical Cabinet Fires by Cabinet Type**

The most interesting insights from this review are that most countable fires originate from sources that have a low volumetric fuel density factor and that motor control centers fires are the most frequent. Of the 68 countable cabinet fires in the past twenty years, 26 of them (approximately 38%) have occurred in motor control centers.

A revision of fire ignition frequencies is well underway and expected to be published in mid-2014. Further evolution of the fire ignition frequencies should consider splitting motor control centers into a separate bin for fire ignition frequency due to the high occurrence of such fires for future estimations of ignition frequencies.

The review of bin 15 fires suggest that 76% of the fires occurred on cabinets containing a low volumetric fuel density. The conclusions of EPRI 1022993 [3], a re-interpretation of existing cabinet fire test data, indicate that volumetric fuel density affects heat release rate (HRR). For example, a cabinet containing qualified cable with a high volumetric fuel density, the scoping peak HRR is calculated to be 17 times the cabinet volume (in cubed feet). In comparison, for qualified cable with a low volumetric fuel density, the scoping peak heat release rate is calculated to be 7.65 times the cabinet volume. As a majority of the countable ignition sources are of low volumetric fuel density, future revisions of fire ignition frequency may want to consider splitting ignition sources by relative fuel density.

3.2 Insights into modelling electrical cabinet fires

A closer look at the current methodology versus operating experience reveals a mismatch in both fire size and damage. Previous versions of the FEDB provided limited information on the event. The addition of reference source (event reports, fire reports, LERs) data can assist in understanding more clearly the event timeline and extent of damage.

NUREG/CR-6850 provides for bounding treatment in some areas related to fire intensity and fire placement to limit the need for detailed analysis. Currently, the accepted practice is to use the heat release rates prescribed in Table G-1 [2] to determine damaged equipment and for initial fire modeling. Table G-1 includes 98th percentile HRR values, and these have been used widely in recent fire PRAs. In a limited number of cases, a detailed evaluation considering actual combustible load and detailed fire modeling analysis has been completed. While this approach may provide for a more realistic treatment of the scenarios, these detailed assessments are difficult to implement on a plant-wide level and cost prohibitive to perform for more than a few scenarios.

3.2.1 Insights from Bin 15 (electrical cabinet fires)

The Updated FEDB contains a drop down list to identify the cause of the fire. Based on the fire event, one of fourteen fire causes is selected. At a high level the cause may include electrical failure, overheated materials, explosion, hotwork, personnel error or mechanical error. A review of the countable electrical cabinet (bin 15) fires indicates an overwhelming majority (79%) of the fires are due to electrical failure resulting in overheated materials. The major research question that needs to be answered would be the potential for overheating materials to evolve from the smoldering phase into a fully developed fire.

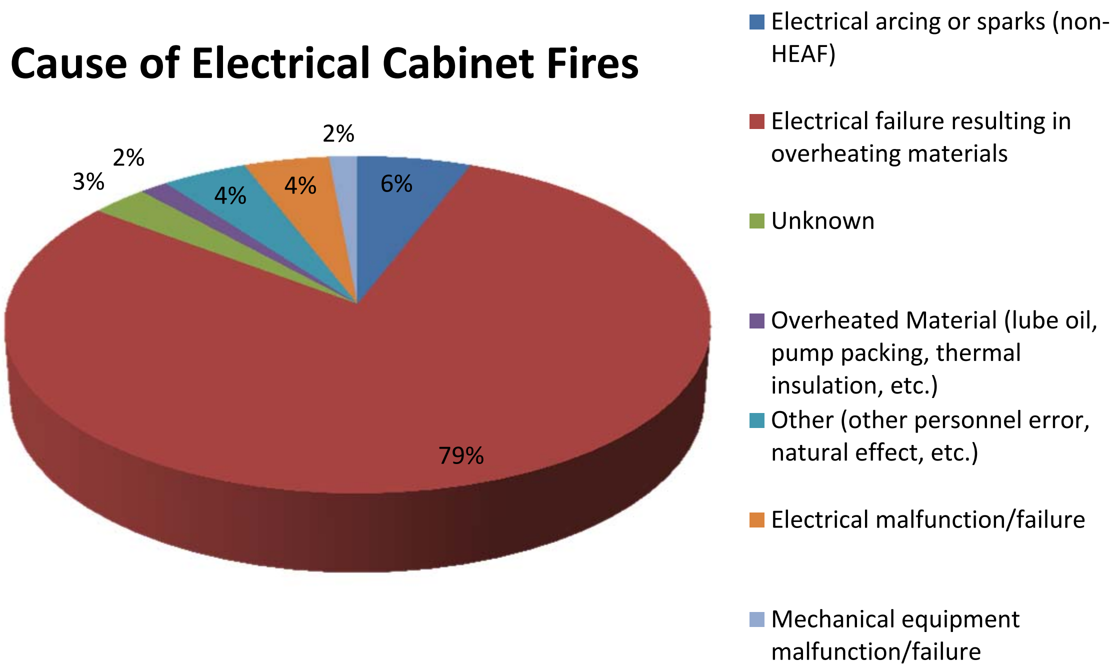


Figure 2. **Cause of Electrical Cabinet Fires**

The remaining causes of electrical cabinet fires are reported in Figure 2. To summarize, the other 20% of fires are initiated as a result of electrical arcing or sparks (6%), electrical malfunction or failure (4%), other (4%), unknown (3%), overheated material (2%) and mechanical equipment failure (2%).

Similar to fire cause, the Updated FEDB contains two dozen items to specify fire type. At a high level the categories include flaming combustion (internal /external to component), smoldering (internal / external to component), overheating, fully developed fire, explosion, arc, no fire, unknown and other. Approximately half of the electrical cabinet fires have evidence of flaming combustion, although most of these fires remain internal to the component. The remaining fire types are reported in Figure 3. It is important to note that none of the cabinet fires caused a fully developed compartment fire.

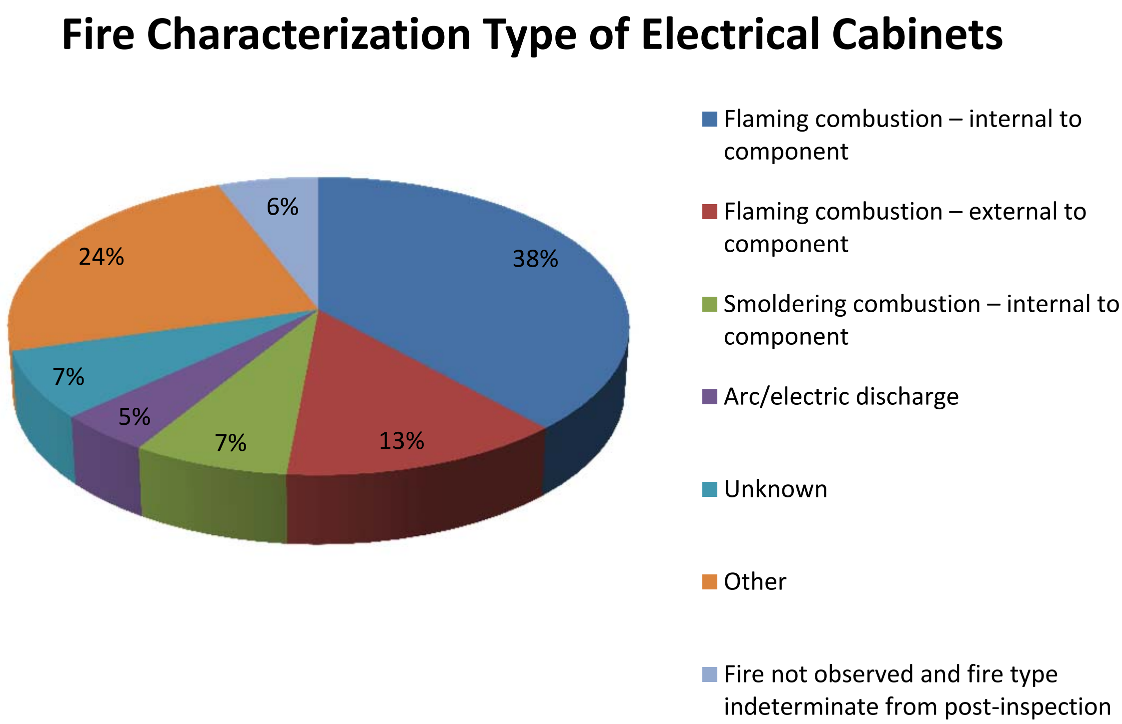


Figure 3. **Fire Characterization Type of Electrical Cabinets**

The Updated FEDB also characterizes fire damage with a drop-down list. The possible choices range from no damage to damage extended beyond structure of origin. For electrical cabinets the most likely options include confined to the object of origin, confined to the object of origin (broad/extensive damage), confined to the object or origin (localized/ single component), and confined to part of a room.

The countable fire events from the past twenty years of operating experience indicate that most fires (54%) are confined to the object of origin (general). When this is expanded to include the more specific object of origin fields (localized/single component and broad/extension damage), this reveals that three out of every four fires do not damage anything beside the initiating component itself. While a not insignificant portion of fires is unknown (15%) and other (24%), this clearly represents a mismatch between operating experience and the current fire PRA methodology. Further research is needed to determine the strength of the ignition source for electrical cabinets as well as to determine the zone of influence outside the initiating component. The current method of assuming that if a fire occurred it would initiate near the top of the cabinet, combined with what appears to be an overestimation of the heat release rate, potentially skews insights and plant risk estimates.

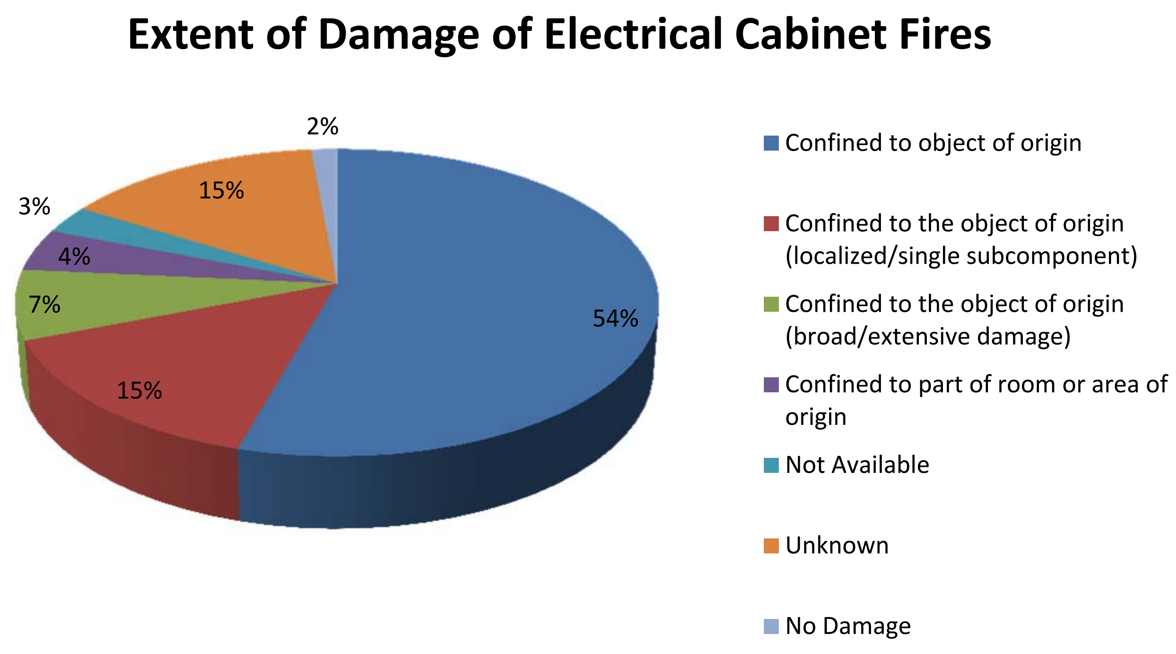


Figure 4. **Extent of Damage of Electrical Cabinet Fires**

The insights provided here utilize the generic categorization of the fields contained in the fire events database. While the information presented in this paper appears to identify a mismatch between operating experience and PRA results obtained using current methods, more detailed research is needed to understand why it is rare to observe large self-sustaining fires in electrical cabinets. EPRI is supporting the joint effort to revise the current methodology by providing additional insights from operating experience beyond the general insights provided in this paper. While this project is just getting underway, this effort will provide additional insights into a variety of relevant aspects:

- Ignition source,
- Incipient fire stage – development from overheating / arcing to fully developed or self-sustaining fire,
- Fire timeline – time to peak HRR and total fire duration,
- Fire detection and suppression,
- Extent of damage, and
- Benchmarking against current modeling assumptions.

This information is expected to support better methods and data for evaluating electrical cabinet fires in the future.

4. Conclusions

The updated EPRI FEDB remains the primary repository of US nuclear power plant fire incident data. With the additional data and reference source traceability, EPRI expects the FEDB to play a more pronounced role in the development of methods and enhancement of data in the future. While this paper focused on the treatment of electrical cabinets, the updated FEDB can be mined for similar insights for other ignition sources, such as transient fires. Current EPRI research is underway to develop more refined treatment for modelling transient fire scenarios.

The updated FEDB can be used to support improved nuclear power plant fire risk analysis. Beyond the fire frequencies and non-suppression probabilities, future enhancements should incorporate fire operating experience where appropriate. The coding and classification of data for each fire scenario was crucial in being able to produce the insights presented in this paper. Additional specific insights will be mined to assist in the development of new data and methods to support the modeling of fire risk in plant risk assessments.

References

[1] EPRI (2013), The Updated Fire Events Database: Description of Content and Fire Event Classification Guidance, EPRI 1025284, 2013.

[2] EPRI and NRC (2005), EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Plant Facilities, Volume 2: Detailed Methodology, NUREG/CR-6850 and EPRI 1011989, 2005.

[3] EPRI (2012), Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires: Reanalysis of Table G-1 of NUREG/CR-6850 and Table G-1 of EPRI 1011989, EPRI 1022993, 2012.

Use and Applicability of the OECD FIRE Database Project

Marina Röwekamp[a*], Matti Lehto[b], Wolfgang Werner[c]
[a] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany
[b] Radiation and Nuclear Safety Authority (STUK), Helsinki, Finland
[c] SAC GmbH, Breitbrunn, Germany

Abstract

Main purpose of the OECD FIRE Database Project is to provide a platform for multiple countries to collaborate and exchange fire event data and thereby, on the one hand to enhance the knowledge of fire phenomena and, on the other hand, to improve the quality of risk assessments requiring fire related data and knowledge. The scope of data collection covers fire event data from all plant operational states including construction and decommissioning phase.

For application within Fire PSA, the FIRE Database is meanwhile principally capable to provide compartment and component specific fire frequencies for different reactor types and plant modes power operation, low power and shutdown, and for the decommissioning phase being different from the fire protection point of view. Moreover, probabilistic applications strive on deriving plant specific as well as generic fire event trees to quantify conditional probabilities of fire induced damage.

In addition, several activities have been conducted or are ongoing for applying the Database for analysis of the operating experience with reportable as well as non-reportable fires in nuclear power plants reported to the OECD FIRE Database. Examples are the analysis of high energy arcing fault induced fires, combinations of fires with other anticipated events and hazards, or the analysis of apparent causes as well as of root causes of fires in nuclear power plants.

1. Introduction and Project History

The OECD FIRE (*F*ire *I*ncidents *R*ecords *E*xchange) Database is one of five databases collecting events from the operation of nuclear power plants (NPP) currently developed under the umbrella of the OECD Nuclear Energy Agency (hereinafter referred to as “OECD/NEA”). Already in the late 1990s it became evident that the only international collection of fire events from NPP provided by the IRS (*I*nternational *R*eporting *S*ystem) is not suitable for specific analysis and use in risk assessment, because only events impacting items important to safety or human health are reported there. In this respect only dedicated databases allow for lessons learned on specific topics as well as for quantitative analysis, in particular within Fire PSA. Therefore such an events database was and is needed.

The purpose of the OECD FIRE Project is therefore to provide a platform for multiple countries to collaborate and exchange fire data and thereby to enhance the knowledge of fire phenomena on the one hand and, on the other hand improve the quality of risk assessments requiring fire related data and knowledge. Applicable to commercially operated nuclear power plants only, the OECD FIRE Database covers a collection of data from fire events from all plant operational modes including construction and decommissioning phases.

After publication of a CSNI State-of-the-Art Report on Level 1 PSA methodology [1], a study on fire risk assessment was started by WGRISK (formerly PWG5), which resulted in the international workshop on fire risk assessment organized by STUK in Helsinki in summer 1999 and in a State of the Art Report (SOAR) on "Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics" [2] published in early 2000. One important concluding remark was the following:

"The shortage of fire analysis data is one of the major deficiencies in the present fire risk assessment. In order to facilitate the situation, it would be highly important to establish an international fire analysis data bank, similar to that set up by OECD for the CCF data collection and processing system (ICDE/CCF data bank at OECD). Such a data bank would provide fire event data on real fire cases, incipient fires (e.g. smoldering) detected/extinguished before development, dangerous or threatening situations, reliability data on fire protection measures, and the unavailability of fire fighting systems, for example, due to component failures or operational errors."

Based on the above mentioned concluding remark, several OECD member countries agreed to establish a project to exchange fire events data to encourage multilateral co-operation in the collection and analysis of data related to fire events in nuclear power plants. During its 2000 annual meeting, the CSNI (*C*ommittee for *S*afety of *N*uclear *I*nstallations) of the OECD/NEA approved the establishment of a Fire Incident Records Exchange (FIRE) Project for collecting data on fire events at NPP in January 2003, which was initially joined by nine countries.

With emphasis on data validity and data quality, OECD FIRE Coding Guidelines for collecting and classifying fire event data have been developed in the first Project Phase (2003 – 2005) to ensure consistent interpretations and applications. Operating Procedures and a Quality Assurance (QA) Manual supplement the Project documentation.

Fire data have been continuously delivered to the FIRE Project since January 2003. Since 2004 and based on the feedback from the first years, stable routines for reporting and QA are in place. The Project was successfully continued with three additional Member countries in Phase Two (2006 - 2009) under a given set of Terms and Conditions. During this Project Phase, several Project members started activities for testing the comprehensiveness of the chosen Database format and its applicability resulting in valuable improvements and retrieving existing information for specific purposes from the Database.

The Member countries of Phase Two of the Project and the Operating Agent (OA) continued this Project in Phase Three (2010 - 2013) under new Terms and Conditions with reasonable progress in applying the Database. This involved answering questions arising in the licensing and in supervisory activities of nuclear power plants in Member states requiring feedback form the fire related operating experience. Other activities were directed at supporting Fire PSA by providing generic, compartment as well as component specific fire frequencies for different reactor types and plant operational states (POS).

The Project is meanwhile being operated in the fourth Project Phase under the auspices of the OECD/NEA. Each of the currently participating Member countries (Canada, Czech Republic, Finland, France, Germany, Japan, Korea, The Netherlands, Spain, Sweden, Switzerland and United States) has nominated a National Co-coordinator (NC) responsible for the administration of the FIRE Project within his/her respective country. In addition, the Project has got an Operating Agent (OA) in charge for tasks defined by NCs.

2. OECD Fire Database Scope and Use

2.1 Project objectives

The OECD FIRE Database Project aims at improving the safety of nuclear power plants by better accounting for feedback from nuclear power plants´ operating experience with fires and by providing common resources for analytical work in the frame of deterministic as well as probabilistic assessment. For meeting this objective, the Project has established a framework for a multi-national co-operation in collecting and analyzing data on fire events in nuclear power plants.

The objectives of the OECD FIRE Project are [3]:

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

The Database is envisioned to be used to

- support fire model development, code validation, etc.,
- identify all types of events and scenarios for inclusion in PSA models ensuring that all mechanisms are accounted for,
- support Fire PSA by real data from NPP operating experience, in particular to evaluate fire occurrence frequencies
- compare national fire event data from member states with the accumulated international data collected within the FIRE Database.

The objectives of the OECD FIRE Database have been further extended during third phase of the Project to cover the following analytical topics identified:

- Further enhancing the Database providing additional guidance for improving narrative fields by prompting questions and event sequence diagrams,
- Grouping events, e.g. "challenging fires", "potentially challenging fires", etc.,
- Performing trending analysis, e.g. for consolidation of national event databases,

- Extending the analysis, e.g. by
  - estimating fire frequencies,
  - quantifying fire scenarios,
  - analyzing human performance,
  - screening out fire scenarios,
  - investigating apparent as well as root causes of fires and related phenomena,
  - analyzing homogenous event groups,
  - estimating fire brigade response times,
  - particularly investigating HEAF (*h*igh *e*nergy *a*rcing *f*ault) events,
  - assessing fire development, growth rate and spreading.

Further extending scope and objectives of the FIRE Database Project is intended and will be discussed within the ongoing fourth phase of the Project. This will cover e.g. extending the Database to other nuclear installations such as larger research and prototype reactors and non-reactor facilities of the nuclear fuel cycle. In the longer term an extension is envisioned of the data collection to data on fire detection and fire suppression systems and equipment that are typically needed in the frame of Fire PSA. An extension to data on events inducing a leak of explosive gas (notably H_2) may also be possible. The data collection may be further extended to better address human factors in Fire PSA.

2.2 Data collection methodology, quality assurance and challenges

With strong emphasis on data validity and data quality, specific OECD FIRE Coding Guidelines (as included in the Database CD [4]) have been developed and continuously improved over the Project phases for collecting and classifying fire event data ensuring consistent interpretations and applications.

The FIRE Project is in principle able to support Fire PSA by providing quantitative insights to be used for fire risk analysis in general. The information on fire events in the Database [4] is meanwhile principally available in a format allowing for quantification of fire specific PSA parameters, such as compartment or component specific fire occurrence frequencies. These data cover different reactor types, plant operational states from construction to decommissioning phases, failures of fire detection and/or extinguishing means. Moreover, the OECD FIRE data can support the generation of generic as well as plant specific fire event trees as outlined in [5].

Fire event data from NPP in countries providing, according to their reporting practice, all fire events to the Database without specific reporting criteria and/or thresholds can be used for estimating fire occurrence frequencies. Together with information on numbers of compartments and components fire frequencies can be estimated for compartment as well as component types. These calculations can be done for all POS, but also separately however for power operational states (referring to more than 5 % of full power level).or low power and shutdown states.

At the time being, the collection of accumulated generic information (e.g. number of components and/or compartments per reactor type and/or plant operational state, number of plant operating years, etc.) is ongoing. As soon as all this information is available it will be provided to all Project Members in anonymous generic form. Completion is foreseen by the end of 2015.

One challenge in setting up an international fire events database is to ensure a consistent reporting level between countries in order to capture all events meeting the objectives of the Project.

Fire[1] events to be included in the OECD FIRE Database have therefore jointly been defined as follows:

- Any process of combustion characterized by the emission of heat accompanied by (open) flame or smoke or both;
- Rapid combustion spreading in an uncontrolled manner in time and space.

However, in this context, it has to be mentioned that there are still limitations for PSA use of data from the FIRE Database, because a statistically significant amount of data is required to support reliable fire risk assessments (particularly for incipient fires). The following limitations have to be explicitly mentioned:

- Due to inconsistencies in reporting thresholds between member countries the database is small for statistical use. For example, there are differences between the reporting criteria for providing fire event data in the different member countries.
  - Some countries do report all fire events having occurred in nuclear power plants (e.g. Czech Republic, Finland, France and Sweden).
  - Other countries are only able to provide those events to the Database which has to be obligatory reported to the national authorities according to the national reporting criteria in place (e.g. Germany, Japan, and USA). Regulatory and utilities' reporting levels are different between these Member countries (e.g., did it or did it not affect safety equipment, different duration thresholds, etc.), and, in addition, the reporting criteria may have changed over time.

  The users of the FIRE Database are made aware of these differences by an Appendix to the Coding Guideline (cf. [4]) where for each country the reporting criteria are provided in detail.

- Not all project data are readily accessible and available in a format for direct application within PSA.

This gap can only be closed if the existing Database is further extended and inconsistencies can be stepwise excluded. In addition, the level of detail and quality of project data from events way back in the past is sometimes insufficient for application within Fire PSA. However, the quality of data provided to the Database is continuously improving with the number of events provided.

Efforts to improve the Database in this direction have already been started and will be continued in the actually ongoing Project Phase. For events from the past, the Database includes for reference the evolution with time of reporting levels. For future events, one objective is to define a Project reporting level, which will account for the countries' policies while correctly addressing the technical Project objectives.

---

[1] *Note that the term "fire" as used in this context includes incipient fires as well as fully developed fires. Fires shall be included in the database if they are relevant to safety and also if the same type of fire has the potential to be relevant/significant for safety under different boundary conditions (such as different ventilation conditions, other plant operating states (POS), same components affected in other locations, etc.). Explosions not resulting in an open flame shall be excluded*

At the time being, the collection of accumulated generic information, e.g. number of components/compartments per reactor type and/or plant operational state, number of plant operating years, etc., is still ongoing and will be provided to FIRE Participants in anonymous generic form. In particular, the Database entries for each fire event cover the following information according to the Coding Guidelines included in [4]:

- General event data (e.g. event title, plant identification (anonymized), registrar, data and time of detection, date of event description and potential revision, FIRE event description, event sequence and interpretation, operation mode prior to the event, confirmation and suppression time, anonymous figures/photographs if appropriate));
- Data on the fire ignition phase (e.g. building and room/plant area where the fire started, type of room and component where the fire occurred, ignition mechanism, root cause(s), type of fire detection, detector type, detection system performance, combustibles/fire loads involved);
- Data on the fire suppression phase (e.g. type of fire extinguishing, extinguishing equipment performance (fixed as well as portable), (manual) fire fighting performance);
- Data on fire effects, other consequences and corrective actions (e.g. operation mode due to the fire, fire influence/effects due to heat, hot gases, or pressure build-up or due to consequential effects on component functions, smoke influence, secondary effects, impact to safety trains, corrective actions);
- References used (names of the reports are provided separately from the Database).

In addition, appendices of [4] provide a glossary of technical terms, and information on reporting levels and thresholds in the participating member countries.

The FIRE Database also contains fire event analysis support data. The *Reporting Threshold Module* defines thresholds for reporting fire events:

- LER (Licensee Event Report) level fires,
- all fires (only some member countries).

The *Fire Brigade Organisation Module* contains the general description of on-site and off-site fire brigade organisation.

2.3 FIRE project participation

Since the OECD FIRE Database Project is operated under the auspices of the OECD/NEA, the OECD/NEA Secretariat has nominated a technical secretariat being responsible for the management of the Project.

Each Member country nominates a National Coordinator (NC) being responsible for the administration of the FIRE Project within the respective country. All National Coordinators together constitute the OECD FIRE Project Review Group (PRG). Technical support to the PRG is provided by the technical secretariat. The PRG does convene regularly on an as required basis, but not less than once per year. Clearly defined PRG rules of procedure are set out in the Operating Procedure (OP).

Coding Guidelines (CG) and a Quality Assurance (QA) Manual (details see [4]) describing the Database framework and data input needs, have been developed within the PRG and may be revised within that same forum. In the event of any inconsistency between the CG or the QA Manual and the Terms and Conditions, the latter will prevail.

Each participating Member is required by the Project Terms and Conditions [3] to submit data on fire events that have occurred in his country through its NC. The data are being collected in the FIRE Database and are being entered according to a coding format specifically developed for this Database, which is explained in the CG and the QA Manual. The Database is to be updated regularly and is designed to facilitate searches.

2.4 FIRE data confidentiality and access

The agreements on data confidentiality and access are fixed in the Terms and Conditions of the FIRE Project (cf. [3]). The purpose of the confidentiality agreement is to protect the data in general against being used in an inappropriate way and keeping the data anonymous, in particular for those member countries not reporting all, but only those fire events having been reported according to national reporting criteria and thresholds,.

Each participant is exclusively responsible for its use of information generated under the Project. It is the responsibility of each NC to identify proprietary information supplied by his/her respective country as such and to ensure that it is appropriately marked. Such information, when included in the FIRE Database, is password protected and accessible only to the Participants, provided however, that nothing in the Terms and Conditions will in any way restrict the owner of that data from disclosing or distributing it to whomever he wishes.

When a NC accesses any data from the FIRE Database that has been provided by another Participant, that NC will mark such data as "*Confidential OECD FIRE Project*" and may not disclose or disseminate that data outside of his/her organisation except that any such data may be disclosed to any other entity with the prior consent of the PRG. Any such data that does not allow the identification of the nuclear power plants may be disclosed to any other entity with the prior notice to the PRG, provided that in each case an appropriate non-disclosure agreement is first entered into between the Participant whose NC is disseminating the data and the entity which is to receive the data.

Any publication or paper discussing the data and/or findings of the FIRE Project will be submitted to the PRG for approval before distribution.

Although in general the Database is accessible to FIRE Project Participants only, a variety of events included has also been publicly reported on an international basis. Information on these events is also available by Non-participants to the Project. In addition, all information published is available also to non-participating PSA developers such that they may benefit from Project activities.

2.5 FIRE database project resources and commitment

According to the Terms and Agreement (cf. [3]), the FIRE Project is completely funded by the Participants with the funding being equally shared amongst participating Member countries. Participation fees are to be paid to the OECD/NEA for reimbursement of the costs incurred by the Operating Agent and the Secretariat, it being agreed that the NEA has a right to receive a moderate administrative fee for its services in an amount to be decided by the PRG.

The schedule for payment of contributions is determined by the PRG. Contributions from Participants due under the schedule have to be paid in full, on the dates specified. The budget for the actual Phase

Four of the OECD FIRE Project is based upon a fee of 12,000 Euros per Member country for the actually two-year period corresponding to a yearly fee of 6,000 Euros per Member country.

New Participants acceding to the Project will be required to pay an entrance fee of 10,000 Euros plus a participation fee equal to the total of the participation fee that such Participant would have had to pay if it had joined the Project at the beginning of Phase Four, unless decided otherwise by the PRG. Thereafter, that Participant will be required to pay the annual participation fee. Extra funding generated from the accession of new Participants will be managed by the PRG.

Within the total operating budget, the OA and the PRG jointly define specific tasks (e.g. Database management, updating of Project documentation, and quality assurance of submitted data) and elaborate the budget which corresponds to each task. The OA has to document its activities for each task to allow approval by the PRG.

The OECD/NEA Secretariat provides secretariat and administrative services in connection with the funding of the Project such as calling for entrance or participation fees and paying expenses to the OA, preparing overall budgets, keeping the financial accounts of the Project and submitting them to the PRG. The OECD/NEA Secretariat does also provide support for the web interface with the Database.

Each Participant has to bear all costs of its participation in the Project other than common costs funded by the Project budget. Withdrawal of a Participant from the Project does not entitle that Participant to any reimbursement of its entrance fee or participation fee paid.

3. Exemplary Applications of the FIRE Database

In the following, a list of topics either already dealt with in the frame of the OECD FIRE Database Project or proposed by PRG members to be analyzed in the frame of the Project is provided:

- High Energy Arcing Fault (HEAF) fire events:

  Operating experience from nuclear installations has shown a non-negligible number of reportable events with non-chemical explosions and rapid fires resulting from high energy arcing faults (HEAF) in high or medium voltage equipment such as circuit breakers and switchgears. Such electric arcs have led in some events to partly significant consequences to the environment of these components exceeding typical fire effects. Investigations of this type of events have indicated failures of fire barriers and their elements as well as of fire protection features due to pressure build-up in electric cabinets, transformers and/or compartments.

  Due to the high safety significance and importance to nuclear regulators, a Topical Report on “Analysis of High Energy Arcing Fault (HEAF) Fire Events” [6] was prepared by an international group that can pool international knowledge and research mean to examine these phenomena in nuclear power plants in more detail for a better understanding of the risk of fire at a nuclear power plants.

  From the safety point of view items important to safety can be damaged severely by HEAF events so that their intended function may be degraded or lost. Such events can produce an arc blast, lead to significant pressure waves and/or cause high speed metal projectiles all of which may impair or damage systems, structures or components (SSC). Therefore, the objectives of the study resulting in the above mentioned Topical Report [6] were to investigate HEAF fire events in the OECD FIRE Database that have indicated failures of fire barriers and their elements as well as of fire protection features due to pressure build-up in electric cabinets, transformers and/or compartments. It was examined if HEAF is a common phenomenon and how it develops, in order to extend the existing knowledge of this particular fire phenomenon, and to improve electrical safety standards and to design proper preventive measures. Last but not least it was to be determined if these events provide sufficient insights on HEAF related events to prevent them and to provide a better understanding of their causes.

Although meaningful statistical conclusions could not be obtained due to the small size of the HEAF event sample, the Topical Report [6] provided the following overall conclusions:

- The 48 HEAF fire events constituted a significant share of more than 10 % of the entire events stored in the FIRE Database.
- The conditional probability of a HEAF event with fire to occur was roughly estimated to be approx. 8 E-03 per reactor year for power operation and 2 E-02 for low power and shutdown states
- In general, there was nothing conspicuous about the occurrence dates of HEAF induced fire events except events at high voltage transformers, for which the occurrence dates seem to suggest a trend of increasing frequencies of HEAF in the more recent years. This might be attributed to ageing problems of electric isolation materials.
- The dominating contribution to HEAF events with consequential fires comes from transformers, with nearly 30 % from high voltage transformers and approx. 8 % from medium and low voltage transformers. HEAF in high voltage transformers typically lead to the destruction of the transformer and consequentially to massive economic losses. Safety significant consequences have not been observed in the events collected for high voltage transformers. The reason for this seems to be the fact that most of these transformers are located outside of buildings or plant areas relevant to safety. On the other hand, HEAF events in high or medium voltage electrical cabinet have a high potential for impairing nuclear safety.
- With respect to the root causes of the HEAF fire events, technical causes (equipment failures) dominated the root causes.

The results of the Topical Report [6] created within the FIRE Project resulted in an experimental research project named "Joint Analysis of Arc Faults (Joan of ARC) OECD International Testing Program for High Energy Arc Faults (HEAF)" under the auspices of OECD/NEA, which is ongoing at the time being.

- Combinations of fires and other hazards, such as seismic, flooding, or explosions:

  Another analytical application with regard to topics mentioned above, such as combinations of events is ongoing. Combinations of hazards, with either a causal relationship or that occur independently, have been systematically addressed and are also covered within the PSA framework as a lesson learned from the Fukushima Dai-ichi reactor accidents. This was the reason for the decision to investigate such event combinations in more detail, starting from a list of possible combinations having been created, searching in the OECD FIRE Database for reported events of this type and analyzing the identified event combinations in more detail. First results on the basis of the OECD FIRE Database in its 2013 version have already been derived depicting that about 10 % of the events in the Database are combination of events. The Topical Report planned to be issued in 2014 will be based on data in the most recent version of the Database available [4], investigating aspects such as components involved in the fire, changes of the plant operational state, and consequences of the event for the plant state.

- Comparison of Fire Protection Standards in Member Countries:

  Another ongoing activity is a Topical Report on "Comparison of Fire Protection Standards in Member Countries" to be prepared under the leadership of Switzerland and the United States to be published during Phase Four of the FIRE Project.

- Application of OECD FIRE Data in fire event tree analyses:

  First attempts for using data from the FIRE Database have been made on a national basis. One example is the application of the Database for plant specific fire event trees. A key element of performing Fire PSA is to determine fire induced equipment failure probabilities for relevant fire sources usually through the use of fire event trees. The Fire PSA analyst derives specific fire event trees for all possible fire sequences considering plant characteristics, compartment specific situations and boundary conditions, potential fire sources and safety targets. Generic fire event trees may be useful for the analyst, however have to be adapted within a plant specific Fire PSA, e.g. branch points should reflect plant characteristics, and branch point probabilities have to be estimated applying plant specific data. Within an ongoing research and development project in Germany [7], a set of generic fire event trees has been developed, consisting of: a time dependent event tree which subdivides a fire event into different phases, an event tree specifically addressing fire detection, and an event tree specifically addressing fire suppression. The above mentioned set of generic fire event trees can be used to analyze the meanwhile more than 420 fire events from 146 nuclear power reactor units. The triplet of sequence numbers for the phases represents an additional attribute of the fire event occurred, which will be stored in the OECD FIRE Database as additional information. The paper presents examples how to use this new attribute of the OECD FIRE Database to retrieve additional information on trends of the fire events observed, which may be used to solve future fire analysis tasks.

- Apparent causes analysis:

  The interest in this topic is to gain further insights into the causes of fire events stored in the OECD FIRE Database and to gain as much knowledge as possible regarding fire phenomena, thereby improving the existing knowledge in fire management. In the frame of the analysis of fire events causes it is interesting to identify if there are any particular characteristics, any patterns that can lead to fire events.

- Challenging fires in areas relevant to safety, such as switchgear fires, relay room fires, MCR fires:

  National applications in Finland were to identify challenging fires of pumps and big transformers for supporting Fire PSA reviews. Mainly, fire propagation and fire impact on neighbouring structures and components were studied based on the event descriptions. At the time when the analyses were carried out, a non-negligible number of 26 fire events occurring at pumps and 23 transformer fires were identified in the OECD FIRE Database.

- Fire suppression analysis
- Rare events
- Database use in front of the background of modernization projects and changes in regulations.

4. Fire Frequency Estimations

Further activities are ongoing to improve the determination of frequencies from fire event data. In this context, the FIRE Database has been re-structured to provide easier to use search capability to support statistical use of fire event data. A screenshot of the most recent version of the Database [4] for event search is shown in Figure 1. Associated nuclear plant operational states (POS) have also been included (summarized by reactor type and country) to allow differentiation between fires occurring during power operation, low power and shutdown states, and decommissioning activities.

Based on reported operating times for the different plant operational states, it is possible to calculate compartment specific as well as component specific fire occurrence frequencies.

As an example, the screenshot from the Database as given in Figure 3 shows those buildings and compartments currently included in the FIRE Database and the respective numbers of fire occurrences for all PWR units for power operation. The corresponding fire occurrence frequencies can be easily determined using the country and reactor type specific operation times for the different POS included in the Database. In order to estimate generic compartment specific fire frequencies average numbers of compartments have to be known. This is at the time being the case for those buildings/compartments and countries outlined in Figure 3. For this subset, Table 1 shows calculated average numbers of compartments and the correspondingly estimated fire frequencies representing approx. 923 reactor years of operating experience.

Regarding component fire frequencies, for some selected important components data on their quantities are available enabling the analysts to derive generic component specific fire frequencies. An example is given in Table 2 for Finland, Germany and Sweden as those countries having already provided sufficient information on component numbers the average numbers of components where fires can occur according to the Coding Guideline [4], providing numbers of fires that have occurred and the correspondingly estimated component specific fire frequencies. In this context, it has to be mentioned that cable specific fire occurrence frequencies are still difficult to estimate due to differences in the approaches used to account for cables across the FIRE member countries.

However, using the reported fire event data, it is meanwhile principally possible to search event data by plant operational state, reactor type, and member country and generate fire occurrence frequencies. In this way, the FIRE database is capable of supporting the application of the fire event data in the frame of Fire PSA.

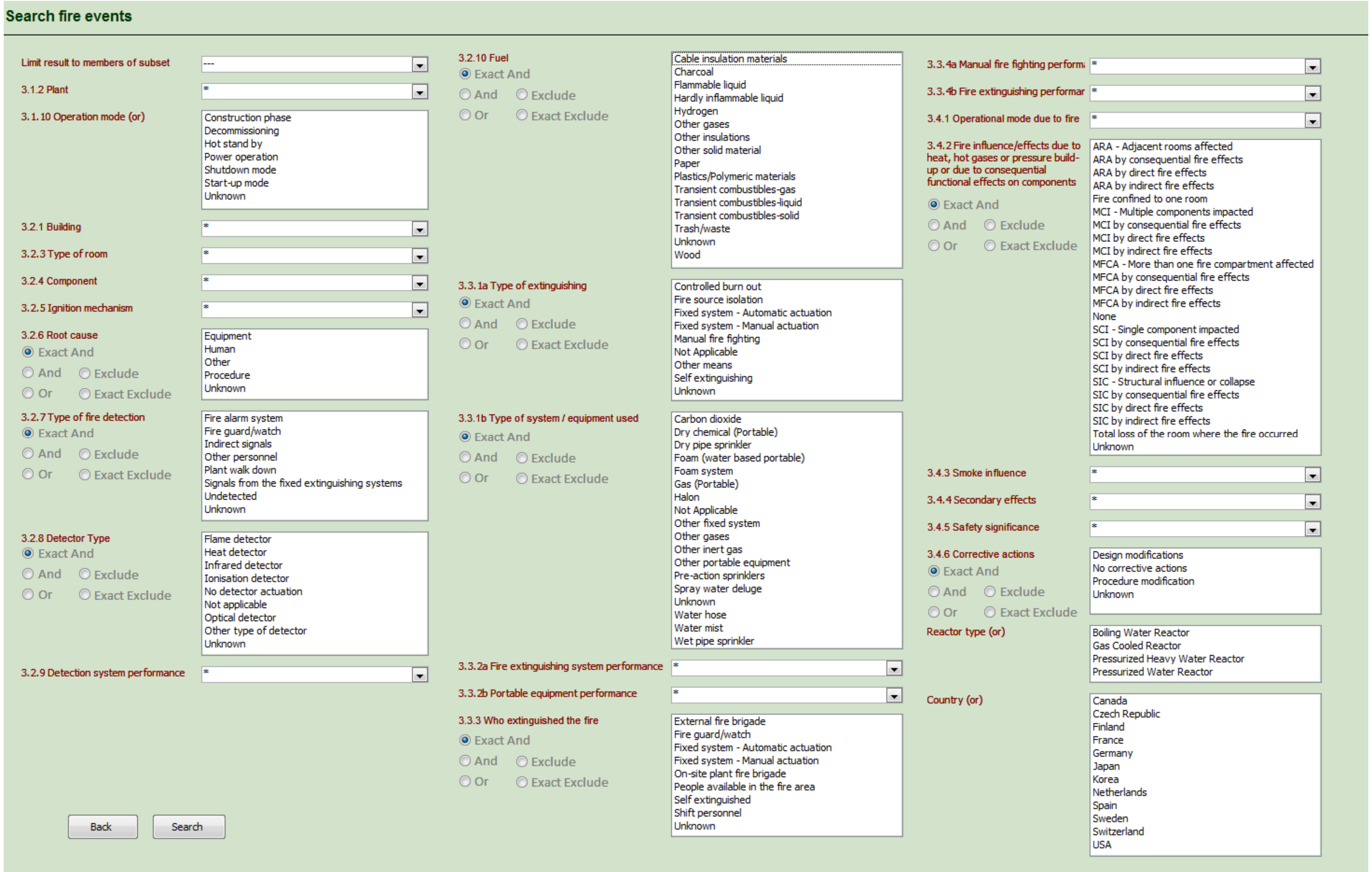


Figure 1. **OECD FIRE Database screenshot for event search**

	Storage for combustible	Staircases and corridors	Diesel generator room	Room for electrical cont	MCR	Other cable room	Office	Other type of room	Storage for other waste	Process room	Switchgear room	Switchyard	Transformer room/bunk	Room for ventilations	Workshop
Auxiliary building		2		1				2	1	7	5			3	1
Diesel generator building				1				1							
Electrical building				10		1		1		1	5			2	
Independent emergency bu											2				
Intake building										3					
Other building/area						1	2	3					2		
Outside the plant (not switc	1							7	1				4		
Reactor building					2					3					
Spent fuel building	1							1						3	
Switch yard								2				4	1		
Turbine building				3		1		12		16	2				

Figure 2. **Number of fire occurrences in PWR buildings/compartments at full power operation (all countries, Fire Database screenshot, from [4])**

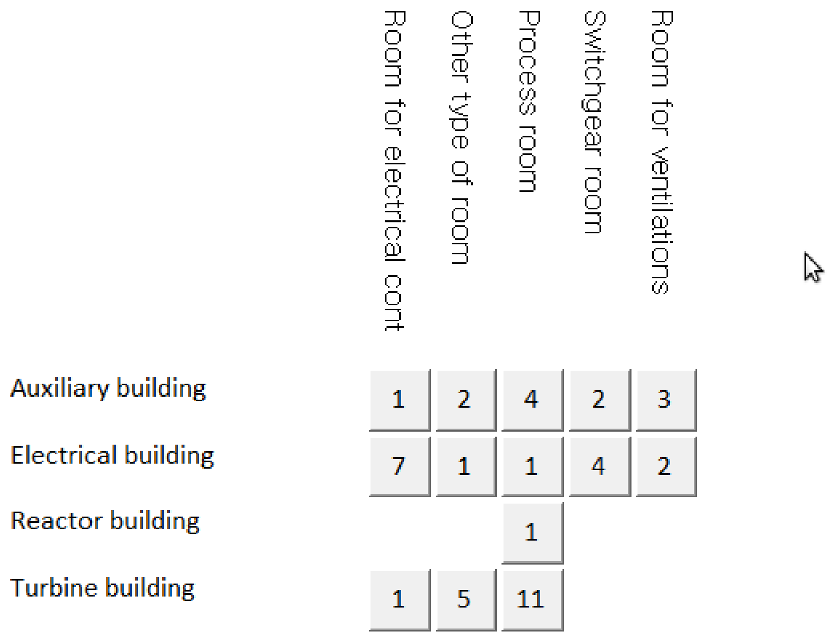


Figure 3. **Number of fires in selected buildings/compartments from countries reporting all events for PWRs with known average compartment numbers (Screenshot from FIRE Database [4])**

Table 1. **Average number of compartments in selected buildings and corresponding fire frequencies for PWR units (power operation, countries reporting all events)**

Compartment Type / Building	Process rooms	Rooms for electrical control equipment. (including main control room)	Rooms for ventilation	Other types of rooms (including staircases and corridors)	Switchgear rooms
Turbine Building	45 2.6 E-04 /a	8 1.3 E-04 /a	7 - *	51 1.0 E-04 /a	4 - *
Auxiliary Building	34 1.2 E-04 /a	4 2.7 E-04 /a	40 8.1 E-05 /a	45 4.9 E-05 /a	9 2.4 E-04 /a
Reactor Building	32 3.4 E-05 /a	16 - *	6 - *	11 - *	12 - *
Electrical Building	18 6.0 E-05 /a	22 3.4 E-04 /a	9 2.4 E-04 /a	14 7.7 E-05 /a	5 8.6 E-04 /a

* no event occurred, therefore no frequency estimate given

Table 2**. Average numbers of components, corresponding numbers of fire events and component specific fire occurrence frequencies for selected components (from those countries having already provided component numbers, all reactor types, power operation)**

Component type	**Average number of components per plant**	**Number of fires**	**Estimated fire frequency[1/a] per component**
High voltage transformer	6.90	3	3.3 E-04
Turbine generator	1.06	8	5.8 E-03
Diesel generator	3.73	2	4.2 E-04
Medium and low voltage transformer	41.20	3	5.6 E0-5
High or medium voltage electrical cabinet (> 1 kV)	1436	14	7.0 E-06
Low voltage electrical cabinet (< 1 kV)			
Electrically driven pump	266	3	8.7 E-06
Rectifiers and inverters	46.26	2	3.3 E-05
Heater	473	4	6.5 E-06
Fan	196	7	2.7 E-05
Battery	28	0	

5. Conclusions

The OECD FIRE Database [4] meanwhile represents a valuable tool for facilitating the use of fire experience of nuclear power plants in 12 OECD Member countries for FIRE PSA. At the end of 2013, the FIRE Database [1] contained more than 420 fire events, the wide majority of them quality assured. The events are from the period from the early 1980s to 2013, with the bulk of the events in the period of the mid-1990s to the end of 2013. Although the reporting of events is not yet exhaustive, the Database provides a good platform for starting the analytical phase.

A variety of applications has already been started making increased use of the newly added enhanced capabilities for interactive queries and evaluation tasks as outlined in Sections 3 and 4.

It is now possible to quickly estimate – according to the analytical task to be performed – fire frequencies for different samples of fire events for all plant operational states, different types of reactors, selected sets of countries under consideration of reporting criteria and thresholds in Member countries. A variety of applications has already been started making more and more use of the enhanced statistical possibilities.

Data collection is continuously ongoing with in average approx. 30 events to be expected per year to be included. This may increase significantly, if larger amounts of fire event data from the U.S. will be submitted in the future.

Acknowledgements

The authors are grateful for the support of the participating countries to the joint OECD FIRE Database Project.

References

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) (1993), State of the Art of Level 1 PSA Methodology, NEA/CSNI/R(92)18, Paris, France, January 1993

[2] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) (2000), Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics - State of the Art Report (SOAR), NEA/CSNI/R(1999)27, Paris, France, 2000.

[3] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) (2014), OECD/NEA Fire Incident Records Exchange (OECD FIRE) Terms and Conditions for Project Operation, Phase Four, 2014 – 2015, Paris, France, 2014

[4] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) (2014), OECD FIRE Database, OECD FIRE DB 2013:1, Paris, France, April 2014

[5] Tuerschmann, M., W. Werner, and M. Roewekamp (2012), Application of OECD FIRE Data for Plant Specific Fire Event Trees, in: 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012 (PSAM11 ESREL 2012), ISBN: 978-1-62276-436-5**,** Curran Associates, Inc., Red Hook, NY, 2012, pp. 3568-3379

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

[7] Einarsson, S., M. Tuerschmann, and M. Roewekamp (2014), Event Tree Methodology as Analytical Tool for Fire Events, in: 12th International Probabilistic Safety Assessment and Management Conference (PSAM12), Honolulu, Hawaii, USA, to be published, June 2014

First Investigations Regarding Combinations of Fires and Other Events in the OECD FIRE Database

Heinz Peter Berg[a], **Nicole Fritze**[a], **Diego Escrig Forano**[b]
[a]Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany
[b]Consejo de Seguridad Nuclear (CSN), Madrid, Spain

Abstract

Operating experience from nuclear installations has shown that combinations of fires and other anticipated events do occur during the entire lifetime of these installations. The required function of structures, systems and components (SSC) important to safety may be impaired in case of the occurrence of such event combinations. This may result in degradation or loss of their intended functions.

Therefore, it was decided to investigate combination of fires and other events or hazards. For this investigation, three types of combinations have to be distinguished: fire and consequential event, event and consequential fire, and fire and independent event occurring simultaneously.

For each of these groups of event combinations it has to be systematically checked, which types of internal or external hazards can be correlated to fire events. The general answer to this question is that only internal hazards may occur as a consequence of a plant internal fire, while fires may be induced by several internal or external hazards. This consideration revealed a list of possible combinations, only some of them have been observed in the operating experience reported to the OECD FIRE Database.

Basis for such a first investigation is the OECD FIRE Database in the version of December 2012 containing in total 415 fire events.

45 out of these 415 fire events have been identified as event combinations of fires and other events in the recent version of the OECD FIRE Database. This contribution of approx. 10.8 % is rather small, however non-negligible. However, it underlines the necessity of design- and site-specific considerations within a Fire PRA including traceable arguments to screen out some of these event combinations. In several cases the plant state changed from full power to low power and/or shutdown as a result of the event combination, in some cases safety trains were lost during the event sequence.

The vast majority of events combinations in the recent version of the FIRE Database are consequential fires after an explosion with 24 events. Ten fire events resulted in an internal flooding, mostly due to the necessary fire extinguishing activities.

The current status of the investigations and their relevance for probabilistic safety assessments is presented.

1. Introduction

Operating experience from nuclear installations has shown that combinations of fires and other anticipated events, in particular external and internal hazards, do occur during the entire lifetime of these installations.

In the past the OECD FIRE ((*F*ire *I*ncidents *R*ecords *E*xchange) Database has mainly be applied for specific questions regarding operational experience with a certain type of fire such as high energy arcing fault fires [1] or specific components which were involved in a fire or where the fire started, e.g. transformer fires [2], [3].

The required function of structures, systems or components (SSC) relevant to nuclear safety may be impaired in case of the occurrence of event combinations of fires and events. This may result in degradation or loss of their intended functions. In principle, combinations of events such as earthquakes and consequential fires may significantly impair or even totally disable SSCs and are often not limited to one reactor unit at multi-unit sites.

In light of that background recent documents of the International Atomic Energy Agency (IAEA) address this topic. For the design of nuclear power plants [4] it is required: "*Where the results of engineering judgment, deterministic safety assessments and probabilistic safety assessments indicate that combinations of events could lead to anticipated operational occurrences or to accident conditions, such combinations of events shall be considered to be design basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence. Certain events might be consequences of other events, such as a flood following an earthquake. Such consequential effects shall be considered to be part of the original postulated initiating event.*"

The IAEA Safety Guide on Level 1 PSA [5] states and recommends: "*Initiating events occurring at the plant may be the result of the impact of a single hazard or a combination of two or more hazards. The possible combinations of hazards should be identified on the basis of the list of individual internal and external hazards. The entire list of potential hazards should be used for this purpose before any screening analysis is carried out. The general approach used for the identification of a realistic set of combinations of hazards should be based on a systematic check of the dependencies between all internal and external hazards.*"

Therefore, the idea was to investigate in more detail the combinations of fires and other anticipated events using the OECD FIRE Database and to prepare a Topical Report. For that purpose in a first step the latest available version of this Database was used [6].

2. Identification of Combinations of Events

The objectives of the planned Topical Report are to:

- Identify scenarios in the OECD Fire Database where fire events and other anticipated events or hazards occurred either independently from each other or as consequence of each other,
- Investigate these event combinations, also with respect to domino effects (e.g. earthquake – high energy arcing fault – fire),
- Determine if the investigations provide a better understanding of the causes and the interdependencies to prevent such event combinations or limit their consequences to safety in the future.

When investigating combinations of fires and other anticipated events or hazards, three types of combinations have to be distinguished:

- Fire and consequential event,
- Event and consequential fire, and
- Fire and independent event occurring nearly simultaneously.

For each of these groups of event combinations, it has to be systematically checked, which types of internal or external hazards can be correlated to fire events. The general answer to this question is that only internal hazards may occur as a consequence of a plant internal fire, while fires may be induced by several internal or external hazards. Combinations of fires and independently occurring hazards are the rarest combinations; only very few external and internal hazards have to be considered to be significant.

This investigation revealed the following list of possible combinations which is in line with international discussions and also with some recently issued national regulations.

- Fire and consequential event:
  - Fire and consequential fire,
  - Fire and consequential explosion,
  - Fire and consequential (internal) flooding,
  - Failure of electrical, mechanical or pressure confining components with the potential of impairing systems relevant to safety.
- Event and consequential fire:

  Internal hazard and consequential fire:
  - Explosion and consequential fire,
  - High energy failure of electrical, mechanical or pressure confining components with the potential of impairing systems relevant to safety and consequential fire,
  - Flooding and consequential fire.

  Natural external hazard and consequential fire:
  - Earthquake and consequential fire,
  - Weather induced natural hazard and consequential fire,
  - Other natural hazard and consequential fire.

  Man-made external hazard and consequential fire:
  - External fire and consequential fire,
  - External explosion and consequential fire,
  - Aircraft crash and consequential fire,
  - Other man-made hazards and consequential fire.
- Fire and independent event:

  Internal hazard and independent fire:

– Fire and independent fire,

– Explosion and independent fire.

Natural external hazard and consequential fire:

– Earthquake and independent fire,

– Flooding and independent fire.

All these combinations have been identified as basis for the assessment, based on international and national experiences and discussions, also triggered after the Fukushima Dai-ichi reactor accidents. However, only a few of these potential event combinations have been observed to date in the operating experience of nuclear power plants in those member countries participating in the OECD FIRE Project as stored in the recent version of the OECD FIRE Database [5].

The investigations have shown that 45 out of 415 fire events have been identified as event combinations of fires and other events. This contribution of 10.8 % is rather small, however non-negligible. The distribution of the 45 events with respect to the type of combinations is provided in Figure 1.

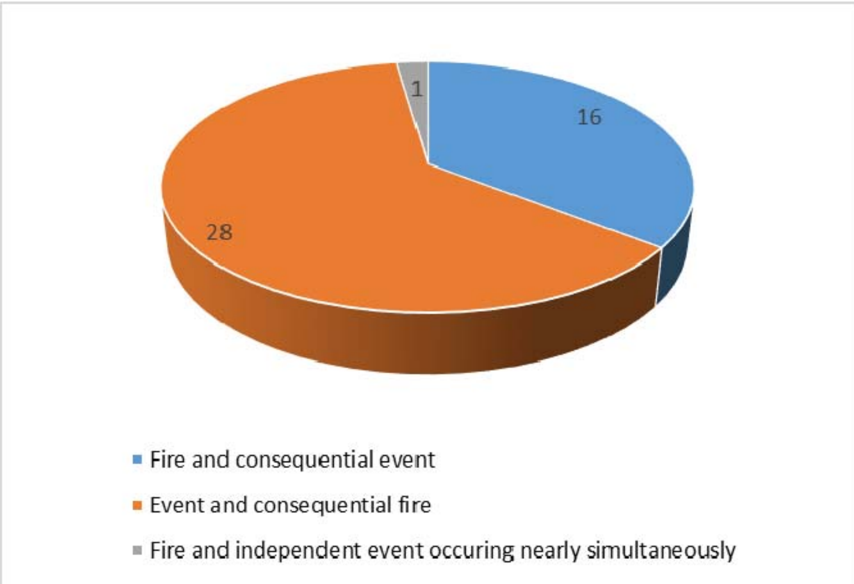


Figure 1. **Types of combinations**

Three of these combinations (approx. 10 % of all event combinations) are combinations of multiple events as in the case of earthquake and a consequential fire (see section 3.2). The aspect to what extent the operational mode has been changed due to the event combination is presented in Figure 2.

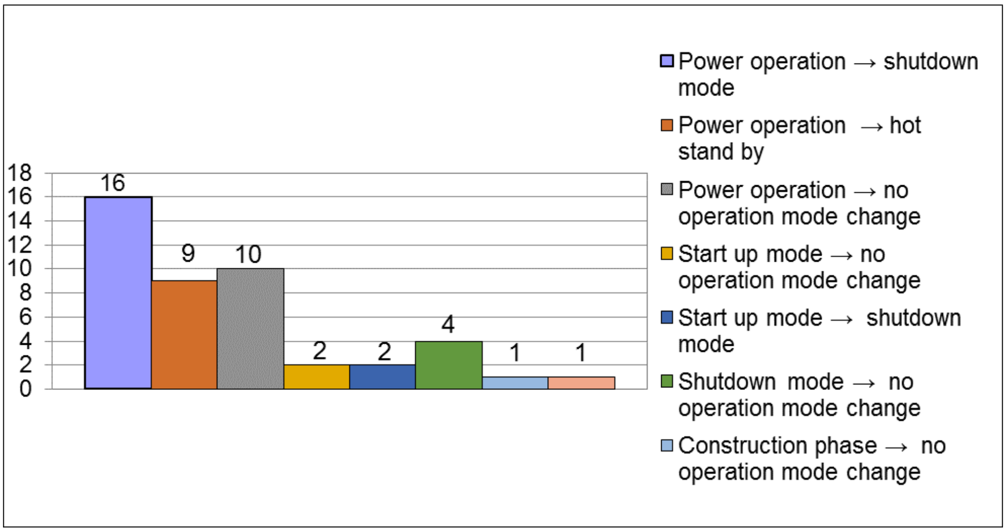


Figure 2. **Change of the operation mode**

3. Examples of Event Combinations Investigated

The first investigation has shown that for most of the different types of combination, only one or very few events have been reported so far. The vast majority of combinations come from explosions and consequential fires with in total 24 reported events representing a contribution of 5.8 % of all fire events in the Database. Thus, the available operating experience gives some indications that this event combination needs special attention with respect to preventive measures.

The analysis of fire and consequential flooding, with a total of ten events representing about 2.5 % of all fire events in the OECD FIRE Database, shows that some improvement could also be made for this type of event combination.

In the following at least one example for each of the three types of combinations

- Fire and consequential event,
- Event and consequential fire, and
- Fire and independent event occurring nearly simultaneously

is provided.

The examples in sections 3.2 and 3.3 are also addressed in [1], because these are high energy arcing fault events which lead to fires.

3.1 Fire and consequential flooding

For the event combination of fire and internal flooding it is remarkable that in the majority of events the water causing the flood comes from the fire protection systems or from the fire brigade. It could be observed that this flooding is caused by one or more of the following factors:

1. Not enough draining capacity in the fire areas,
2. Drainage obstruction: prior to the fire or apparently caused by some “products” of fire,

3. The existence of a water escape route does not seem to be known or analyzed that is an amount of water is a consequence of the fire extinguishing. As an example, a fire burning a joint causing the water to flow from the area where the fire extinguishing started to an adjacent area or compartment can be mentioned.

One possible interpretation is that one of the root causes behind most of these events is a lack of a comprehensive analysis involving the drainage system capacity, and possible flood paths resulting from water coming from the fire protection system causing flooding not predicted. It seems useful in the design of fire protection systems to analyze more carefully what will happen with the fire water from the fire extinguishing system in case of real fire and to establish a management strategy for fire agent effluents.

3.2 Earthquake and consequential fire

Two examples of this type of event combination can be found in the Database. In both cases a sequence of steps of the fire event can be observed for the combination of an earthquake and consequential fire. Not surprisingly the earthquakes have led to a high energy arcing fault which resulted in a fire. Corrective actions have been plant specific design modifications, in one case providing an additional fire engine (chemical) and full time operators for this fire engine.

In the first case, an earthquake occurred and the reactor was automatically tripped from 100 % of full power by high seismic acceleration signal prior to the fire, and was cooled down to the cold-shutdown mode without suffering any effects from the fire. The fire started at the house transformer that was installed outside and isolated from other components by a fire wall. The ignition mechanism was that the electrical arcing between the bushing and the bus duct had ignited the insulation oil leaked from the transformer to the bus duct. The fuel involved was insulation oil leaked from the transformer. The transformer contained about 17 m^3 of insulation oil during normal operation. The fire was detected by post-earthquake patrol of plant personnel. The fire was extinguished by chemical hydrate from the regional fire engine.

In case of the second event, the earthquake caused an arcing fault in two of ten sectors of the non-emergency switchgear cabinet. The arcing fault resulted in a fire affecting all ten sectors within the cabinet. The cabinet was installed in the underground floor of the turbine building. Prior to the earthquake, the plant was at full power operation, it was automatically shut down due to the signal of high seismic acceleration. The fire was detected by an optical detector, although the on-site fire brigade could not identify the fire location due to heavy smoke at first. Additionally, the public fire brigade was called which was not able to support the on-site fire brigade because of damage of the access ways to the site due by the earthquake and tsunami. The fire duration was nearly 8 h, one of the highest fire durations observed from events in the OECD FIRE Database.

3.3 Explosion and consequential fire

As already explained, explosion and consequential fire is the combination with the highest number of all event combinations. In one case, the explosion resulted in a fire and additionally in some missiles. The amount of high energy arcing fault (HEAF) events for this type of event combinations is 13 out of in total 24. Moreover, a loss of safety trains occurred in ten cases, in one case all train were lost (see example described below).

Figure 3 shows the distribution of components where the fire started indicating that transformers are the components providing the dominating contribution of events.

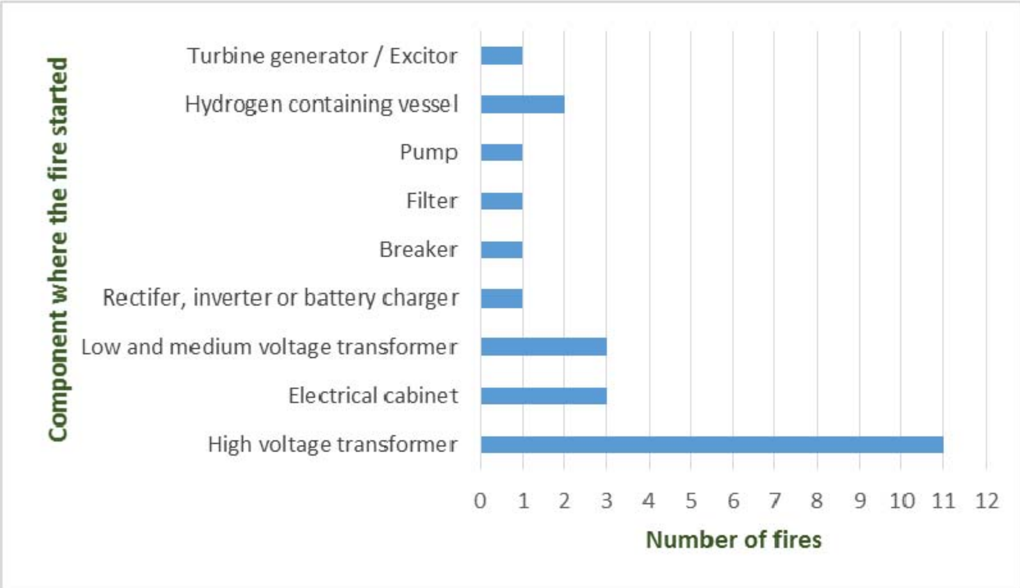


Figure 3. **Component where the fire started**

One example of this type of event combination is a high energy arcing fault (HEAF) fire event in the main transformer. The event combination started with a short circuit in the high voltage terminal (400 kV) due to an isolation failure between this terminal and the phase box. The corresponding high energy arcing event had immediate consequences. The unit was at 100 % power when the main transformer that was installed outside caught fire. The area has got 3 h fire resistant rated walls at both sides because the affected transformer is located between two further transformers. Immediately after the transformer explosion, the turbine trip occurred due to protection actuation and, consequently, reactor trip. The response of plant systems was adequate, particularly transfers of electrical buses and start-up of diesel generator. The plant remained under control in hot stand-by state. The fire protection system of the area was signaled as being activated in the control room. The control room manager ordered visual inspection of the transformers area to the control room. As the plant field operator confirmed the fire, the on-site fire brigade was informed and could in a short time extinguish the fire. Thus, the fire lasted only about 15 minutes.

A further example is an event combination at a multi-unit site with four nuclear power plants. Again, a HEAF event with explosion followed by a fire occurred on the transformer which supplies house load of one plant during full power operation. The transformer failure caused an outage of the 400 kV power line. The unit was then powered from the reserve. The fire fighting was performed by the on-site fire brigade; the fire duration time was about 15 min. The probable cause was the fault of the power contacts of the branch switch of the transformer. HEAF was created before total short circuit. As a result, in a very short time a large volume of gases was generated which caused the explosion of the switch power part and subsequent rupture of the transformer vessel, ejection of oil in the adjacent area, and oil fire. The accident led to the discharge of approximately 32 t of oil from the collecting tank underneath the transformer. This event combination finally resulted in the loss of all safety trains.

3.4 Fire and independently occurring event

The OECD FIRE Database contains only one example for a fire and an independently occurring event, in this case an external flooding. At the time of the internal fire, the plant was in cold shutdown and had already reported the impacts of the flood. Licensee fire brigade personnel as well as personnel of the local fire department responded to the fire. The fire was extinguished after 40 minutes. The fire resulted in a loss of power to six of nine safety-related 480 V ac buses and two of four safety related 4160 V ac buses leading to the loss of the spent fuel pool cooling function. This could have resulted in the loss of a safety function or multiple failures in systems used to mitigate an event in case the event would have occurred at power.

This combination of events shows the need to be aware of the possibility of a fire and an independently occurring event. In case of the external flooding the accessibility of the plant even under such extreme circumstances is necessary and should be assessed in a PSA to ensure technical support from outside, not only to allow the change of the personnel on the plant but in this case also to enable the accessibility for the local fire department.

4. Concluding Remarks and Outlook

45 out of 415 fire events have been identified as event combinations of fires and other events in the version of the OECD FIRE Database covering fire events by the end of 2012 [6]. This contribution is rather small, however non-negligible. All event combinations were limited to one plant unit in case of multi-unit sites.

The investigation has shown that the number of event sequences representing an event combination of the same type is typically only one, two or four. There are only two types of event combinations where the FIRE Database contains significantly more events.

24 out of 45 events combinations are fires consequential to an explosion and ten out of 45 are fire events resulting in an internal flooding because of fire extinguishing activities. Three event combinations show a domino effect (earthquake resulting in a high energy arcing fault and a consequential fire, and fire resulting in an explosion and a consequential fire).

One example of a fire and an independently occurring event (flooding) in the FIRE Database underlines that such combinations do not only represent academic assumptions. In case of 27 of the combined events the plant operational state changed, in 16 cases from full power to low power or shutdown. In case of ten event sequences, as a result of the combination of events, at least one safety train was lost during the course of the event combination sequence; in one case even all safety trains were lost.

Regarding the use of the OECD FIRE Database for PSA there are still limitations due to inconsistencies between member countries according to differing national reporting criteria. However, the data – as indicated in the case of combinations of fires and other anticipated events or hazards – provide insights for probabilistic considerations.

The scope of the Topical Report on event combinations with fires and other events planned to be issued latest in early 2015 will be to identify:

- Equipment/components involved in these event combinations,
- Duration of fire,
- Fire suppression means applied, assessment of their efficiency,
- The plant operational state (POS), when the events occurred, and potential changes of the POS as a result of the events,
- Good practices to efficiently prevent these events in the future, if possible and

- Presentation of national regulations/recommendations addressing event combinations of fires and other events and how to manage them.

Basis for the Topical Report will be the updated version of the OECD FIRE Database containing fire events up to the end of 2013. Lists of the events with details regarding plant state, component where the fire started, fuel involved, plant area affected, root causes, fire suppression means and duration time of the fire will be presented in tables. Some exemplary event combinations will be discussed in more detail. Moreover, existing national regulations regarding event combinations will provided.

References

[1] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) (2014), OECD FIRE Project – Topical Report No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events, Paris, France, NEA/CSNI/R(2013) 6, June 2013

[2] H. P. Berg, N. Fritze, B. Forell, M. Röwekamp (2010), Risk Oriented Insights in Transformer Fires at Nuclear Installations, Proceedings of ESREL Conference, Rhodes, September, 5 -11, 2010, 351 – 361, 2010

[3] H. P. Berg, N. Fritze (2013) Risk and Consequences of Transformer Explosion and Fires in Nuclear Power Plants (2013), Journal of KONBiN 23 (2013), No.1, 5-16

[4] International Atomic Energy Agency (IAEA) (2012), Safety of Nuclear Power Plants: Design, Specific Safety Requirements, IAEA Safety Standards Series No. SSR-2/1, IAEA, Vienna, January 2012

[5] International Atomic Energy Agency (IAEA) (2010), 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

[6] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) (2012), OECD FIRE Database, Version: OECD FIRE DB 2012:1, Paris, France, March 2013

U.S. NRC Fire Safety Research Activities

N. Melly, G. Taylor, and D. Stroup
U.S. Nuclear Regulatory Commission
Washington DC, 20555
Nicholas.Melly@nrc.gov
Gabriel.Taylor@nrc.gov
David.Stroup.nrc.gov

Abstract[1]

The NRC is actively pursuing a number of research activities in the area of fire safety. One of the objectives of the fire research branch is to improve the agency's knowledge in areas where uncertainty exists in support of regulatory decisions for existing or new designs and technologies with respect to fire safety. This paper will provide details related to high priority test programs such as the Electrical Enclosure Heat Release Rate test program, the Joint Analysis of Arc Faults (Joan of Arc) Organization for Economic Co-Operation and Development (OECD) International Testing Program for High Energy Arc Faults and the Evaluation of Incipient Fire Detection System Performance for Fire Probabilistic Risk Assessment. The paper will also discuss current data and techniques used in NUREG/CR-6850 EPRI TR-1011989 "Fire PRA Methodology for Nuclear Power Facilities" and areas that are being improved with the current programs.

1. Electrical Enclosure Heat Release Rate

1.1 Purpose

In 2005, the US Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) jointly published NUREG/CR-6850 (EPRI 1011989), EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [1]. This report documented methods, tools, and data for conducting fire probabilistic risk assessments (PRAs) in commercial nuclear power plant (NPP) applications. The NUREG/CR-6850 (EPRI 1011989) fire PRA methodology consists of 16 separate tasks with multiple steps for each task. In addition, there are two "support tasks" and a number of appendices with supplemental information required for each task. Using a significant body of previous research on material combustibility characteristics , the authors of NUREG/CR-6850 (EPRI 1011989) established a list of recommended heat release rates (HRR) for a variety of potential fuel types including electrical cables, electrical cabinets, flammable liquids, and transient combustibles. The HRR is a measure of the rate at which a burning item releases chemical energy and has often been characterized as the single most important variable in understanding a fire hazard [2].

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

While NUREG/CR-6850 (EPRI 1011989) is a guidance document for fire PRA in NPPs, the HRR information is useful for both fire PRAs and fire hazard analysis (FHA). Over the last several years, the NRC has been working to improve and refine the HRR values presented in NUREG/CR-6850 (EPRI 1011989). Recently completed work to characterize the HRR, flame spread and ignition of electrical cables has been conducted under NRC sponsorship [3]. The current project is focused on better characterizing the peak HRR and fire growth model for electrical enclosures.

Electrical enclosures[2] are found throughout NPPs. These enclosures are typically constructed of metal on the order of 16 gauge steel (1.5 millimeters (mm) or 0.06 inches (in). The geometries range from small wall-mounted cabinets to large vertical electrical enclosures of multiple sections with various ventilation configurations. Electrical components (wires, relays, circuit breakers, transformers, etc.) are installed inside the enclosures and vary in physical size, function, and electrical specifications. Junction boxes, pull boxes, and similar smaller enclosures are not considered in this study.

Fire in electrical enclosures has been identified as a significant contributor to fire risk in NPPs. During an Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting on November 16, 2010, EPRI identified electrical panel fires as a very significant risk driver (see Figure 1) for the licensees making the transition to National Fire Protection Association (NFPA) 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001)” under Title 10 of the Code of Federal Regulations (10 CFR) 50.48(c). The combination of combustible materials and electrical energy within the electrical enclosure can lead to fires and possibly High Energy Arcing Faults (HEAFs). These fires have the potential to disrupt electrical power, instrumentation, and control in the plant. To gain a comprehensive understanding of the fire phenomena within electrical enclosures, the effects of enclosure size, openings, cabinet function, and quantity of combustible material is being examined.

This test program is confirmatory research based on fire physics to quantify the rate of energy release and spread of fires within electrical enclosures. The testing phase of this project will evaluate the potential HRR and fire growth rates for electrical enclosures typically found in NPP. The test data will be used to support the development of improved guidance for understanding and modeling electrical enclosure fires.

[2] *“Enclosure” is defined in IEEE 100, The Authoritative Dictionary of IEEE Standard Terms, as a surrounding case or housing to protect the contained equipment against external conditions and to prevent personnel from accidentally contacting energized parts. “Cabinet” is defined as an enclosure designed either for surface or flush mounting, provided with a frame, mat, or trim in which a swinging door or doors are or may be hung, and housing modules, backplane(s), I/O connector assemblies, internal cables, and other electronic, mechanical, and thermal devices.*

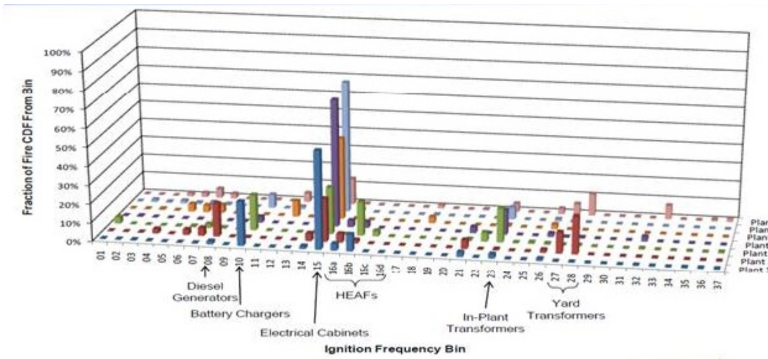


Figure 1. **Fire CDF contribution by ignition source**

1.2 Background

The HRR has typically been characterized as the single most important variable in FHA [2]. HRR is a measure of the rate at which a burning item releases chemical energy. HRR is used as input to a computer fire model and informs fire size, fire growth rate, available egress time, and suppression system impact when all parameters affecting HRR are known [4].

Classically there are two broad categories of methods for estimating the potential impact of burning materials on occupants and structures— risk-based and hazard-based. Both types of methods estimate the potential consequences of possible events. Risk-based methods also analyze the likelihood of scenarios occurring while hazard-based methods typically do not [4]. In a FHA the impact of fire exposure on people and structures is evaluated based on one or more assumed fire scenarios. The HRR of the burning items is a key element of each fire scenario and defines the "size" of the fire exposure [5].

If the heat of combustion of a material is known, the HRR of a burning item can be determined by measuring the item's mass loss rate during combustion (i.e., the mass burning rate). Alternatively, the HRR can be determined directly from measurements of the product of combustion gases collected in an exhaust hood. Laboratory calorimeters can provide useful information concerning HRR using small material specimens. However, these calorimeters may not represent the actual performance of a material when used in the actual built environment. Laboratory calorimeters do not usually test the same size materials as found in most fire scenarios and the laboratory results can be influenced by the relative closeness of the material edges to the center of the flame zone. There is also little or no geometry consideration included in small-scale tests. Intermediate and large-scale calorimeters attempt to provide a more realistic method for understanding actual materials HRR.

In general, combustible materials in NPPs can be divided into four broad categories, including the following [6]:

(1) transient solid and liquid fuels

(2) in situ combustibles consisting of both solid and liquid fuels

(3) liquid fuels used in NPP equipment

(4) explosive and flammable gases

Solid transient fuels include general trash, paper waste, wood, plastics, cloth, and construction materials. Liquid transient fuels commonly include cleaning solvents, paints, and lubricants being used for maintenance of plant equipment. These fuels are generally found in small quantities in most NPP areas at any given time. The most common category of potential fuels found in NPPs is that of in situ solid fuel elements. Of these, the largest single potential fuel source is electrical cable insulation and jacketing materials. These “cables” vary widely in size and location, most commonly installed in open cable trays where they present a hazard as an exposed combustible. Liquid fuels include lubricating and cooling oils, cleaning solvents, and diesel fuels. Finally, explosive and flammable gases can be present in NPPs with hydrogen being the most abundant.

The electrical enclosure HRR project will be to conduct experiments on actual electrical enclosures that are commonly installed in NPPs and were selected according to size and ventilation type to cover a range of plant configurations. The enclosures will be “mocked up” with cabling and components to simulate combustible loading configurations for electrical enclosures identified during the plant site visits conducted by the NRC staff. The ignition conditions necessary to obtain established burning within the enclosures will be examined. The time history of the HRR and the extent of the fire within the electrical enclosures will be measured and documented.

The experimental data obtained will be used to re-evaluate the HRR information contained in NUREG/CR-6850 (EPRI 1011989) and the 2012 EPRI guidance document [7]. Specifically, a joint working group consisting of EPRI and NRC sponsored experts will be assembled to review the available information and develop revisions where appropriate to NUREG/CR-6850 (EPRI 1011989). The NRC expects to use an expert elicitation for portions of this work. The panel will focus efforts on examining cabinet configurations, refining the electrical cabinet frequency bin to reflect plant experience, evaluating the potential to develop a fire hazard model for electric cabinets, and use the electrical enclosure HRR and fire growth information to develop revised HRR distributions for use in PRA applications. A technical report will be issued upon completion of the working group efforts documenting the outcome and associated peer review of the experts’ analysis and conclusions.

1.3 Technical Approach

Staff from the National Institute of Standards and Technology (NIST) and NRC performed site visits to several commercial NPP facilities. These sites included: Bellefonte, Three Mile Island, and Zion. During these visits, site staff members were able to open numerous electrical enclosures to allow the NIST and NRC staff to take photographs and document general characteristics of the enclosures including amount of combustibles and ventilation conditions. From these site visits, the NIST and NRC staffs were able to gain an understanding in the variations between the types of electrical enclosures used in NPPs. The NPP electrical enclosure pictures will be used to inform and develop combustible loadings for the “mocked up” cabinets that are representative of those found in NPPs.

The cabinets will be modified to represent various enclosure configurations that were identified during the plant walkdowns. The combustible loading, i.e., bundles of electrical wire and cables, will typically be described in terms of total mass per unit length, total length, number of conductors, plastic and cooper mass ratio, and whether the cables are used with or without the jacket. The final report will document the relationship between a test cabinet mock-up and the associated actual plant configuration (see Figure 2). The subsequent working group will also use this information to develop guidance for quantify combustible loading.

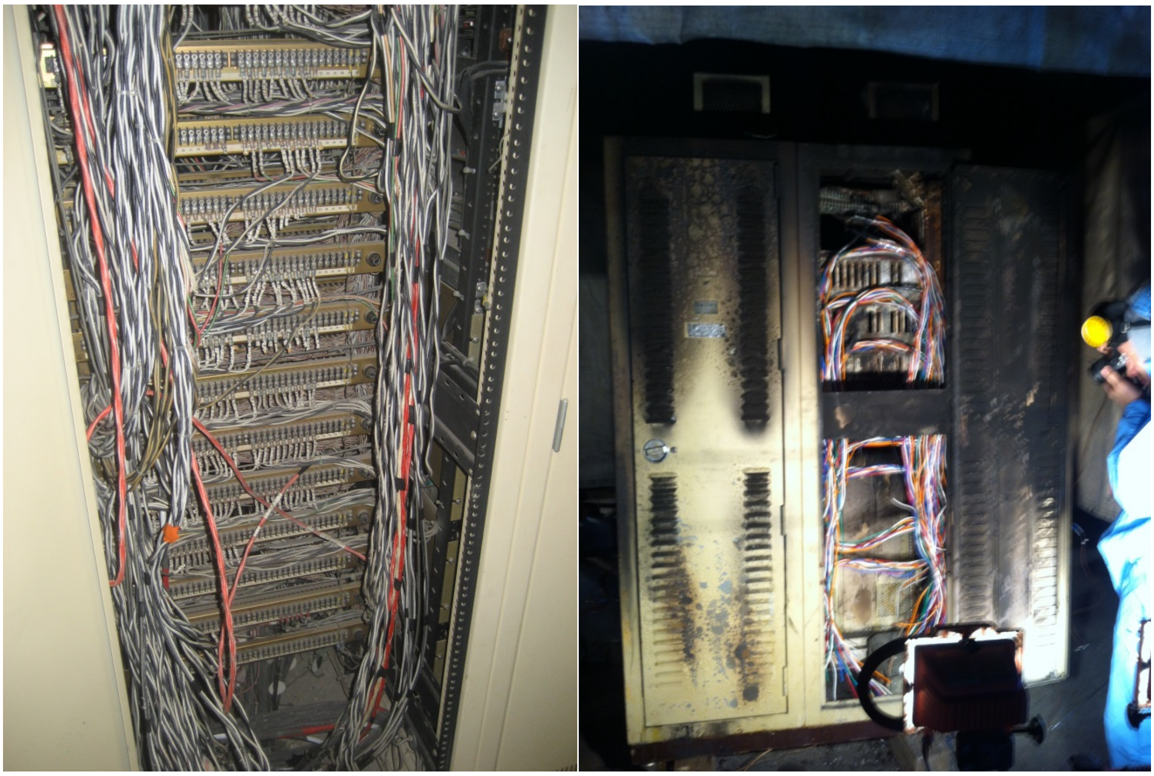


Figure 2. **Actual plant loading example (left), Thermoplastic unjacketed "mock up" (right)**

The primary measurement of the testing is the HRR of the burning electrical enclosures. This measurement is to be achieved by way of oxygen consumption calorimetry. Infrared and conventional video cameras provided by NIST will also be positioned outside of the enclosures. All fire development information such as; time to ignition, steady state, time to peak HRR, decay phase and time to self-extinguishment will all be documented in the final experimental report. Thermocouples are to be positioned inside the enclosure 0.15 meters (m), (6 in) below the ceiling and at 0.3 m (1 foot (ft)) intervals. These thermocouples will be monitored during the tests to indicate if the fire spreads beyond its origin. Combustible load measurements of the cables will be evaluated in terms of length, weight and combustible load based on information obtained though small scale cone calorimetry insights.

Previous testing conducted at Sandia National Laboratories[8][9][10], VTT Technical Research Centre of Finland [11], used either a propane burner or a "transient" ignition source[3] yielding between approximately 1 to 30 kilowatts (kW) to ignite the cable fuels. Based on review of the fire event database, the testing requirements for qualified cable, and preliminary test results, several ignition scenarios have been identified. The ignition scenarios consist of various combinations of a propane burner and a small pan of acetone. The ignition source will simulate a range of ignition energies 1kW—30kW (Figure 3). Ignition will be done in such a way that fires achieve established

[3] *The "transient" ignition source consisted of a 9.5 L polyethylene bucket, with an open 0.5 kg box of Kimwipes, and 0.946 L of acetone in the bucket (NUREG/CR-4527, An Experimental Investigation of Internally Ignited Fires in NPP Control Cabinets, Volume 1: Cabinet Effects Tests, April 1987).*

burning condition and then will be allowed to burn out naturally. After established burning has been achieved the ignition source will be removed allowing for unaided combustion to continue.

A liquid acetone fuel source will be used to simulate the larger ignition energy events (>10 kW) because of safety issues associated with using a propane gas burner in the electrical enclosure which is defined as a confined space. Acetone will also be used for cabinet preheating purposes because of the cold conditions of the laboratory space during this program. Preheating was necessary because the majority of testing was performed in the winter of 2013 when the unheated ambient laboratory conditions were below freezing. Researchers selected acetone to provide a constant HRR source that is easily reproducible throughout the series of test experiments. The goal was to simulate the electrical enclosure at their operating temperature range when the test starts. The burning rate of acetone has been well established by various research studies and is a function of the surface area available for combustion. The HRR and burning duration can be controlled by selection of an appropriately sized container and quantity of acetone. The HRR ignition profiles selected are comparable to ignition methods used in other ASTM standardized fire test procedures such as ASTM E1537, Standard Test Method for Fire Testing of Upholstered Furniture, (propane burner 19.7 kW) and ASTM E1822, and Standard Test Method for Fire Testing of Stacked Chairs, (propane burner 17.8 kW).

Figure 3. **Ignition methods: 1-2kW propane burner (left), 10kW propane burner (middle), 20kW acetone pan (right)**

1.4 Test Results and Fire Model Improvement

After completion of the testing program the NRC will establish a balanced “working group” of experts who will use lessons learned from the application of NUREG/CR-6850 (EPRI 1011989), insights from the testing and their own experience to enhance the guidance currently found in NUREG/CR-6850 (EPRI 1011989). An expert elicitation may be conducted to evaluate all available test data to re-quantify electrical enclosure HRR distributions. These distributions will be developed such that they can be used to enhance Fire PRA guidance. To improve these distributions, the panel will focus on what it believes to be first order parameters impacting HRR such as; fuel load, ventilation, fire growth profiles, fire elevations and fire location within a cabinet.

As a whole, these experts will possess collective expertise in fire testing, fire modeling and fire PRA application. EPRI participation in the working group will provide a balance between the regulator (i.e., two or three members from the NRC and National Laboratories) and EPRI (i.e., two or three members from EPRI/nuclear power industry/consultants). The panel will use test results, previous research, expert judgment and operating experience to develop a new approach to determining electrical enclosure HRR distributions.

2. Joint Analysis of Arc Faults (Joan of ARC) OECD International Testing Program for High Energy Arc Faults

2.1 Purpose

NUREG/CR-6850 Appendix M delineates a High Energy Arc Fault (HEAF) event into two phases: an energetic phase and an ensuing fire. One of the key components in addressing a HEAF event is characterizing the damage during the energetic phase of the event. The enclosure or cabinet in which a HEAF event occurs may be breached by the explosive release of energy. This factor greatly influences the immediate heat flux to which nearby objects are exposed. The ensuing fire is also of interest, as it can be the cause for fire spread and failure of additional components and adjacent equipment that was not damaged by the initial heat released during the HEAF event. One of the major limitations to Appendix M is that it presents a “one size fits all” model. That is, as long as a component meets the criteria for inclusion as a HEAF source, there is no further distinction made based on the characteristics of the specific initiating component (e.g., voltage or current level, device type, etc.) nor those of the component’s enclosure (e.g., robustness of the surrounding cabinet).

This project was identified as part of the OECD fire events database program. Catastrophic failures of energized electrical equipment, referred to as HEAF, have occurred in NPP components throughout the world. HEAF’s typically occur in 480 volt (V) and higher electrical equipment and cause large pressure and temperature increases in the component cabinets, which could ultimately lead to serious equipment failure and secondary fires posing a NPP risk. Most recently the United States has experienced events at Palo Verde in 2013, H.B. Robinson in 2010 and Columbia in 2009. Discussions at the OECD Fire Incidents records exchange meetings indicate similar HEAF events have recently occurred in Canada, France, Germany and most recently at Japan’s Onagawa NPP during the earthquake and tsunami of 2011. OECD Fire Project – Topical Report No.1 “Analysis of High Energy Arcing Fault (HEAF) Fire Events, NEA/CSNI/R[13] published June 2013 documents these international events.

HEAFs have the potential to cause extensive damage to the failed electrical component and distribution system along with adjacent equipment and cables within the zone of influence (ZOI). The significant electrical energy released during a HEAF event can act as an ignition source to other components. HEAF phenomena have been identified as a risk driver and represent a unique challenge for PRA practitioners and fire modelers.

The primary objective of this project is to perform experiments to obtain scientific fire data on the HEAF phenomenon known to occur in NPPs through experiments. The goal is to use the data from these experiments and past events to develop a mechanistic model to account for the failure modes and consequence portions of HEAFs. These experiments are expected to improve the state of knowledge and provide better characterization of HEAF in the fire probabilistic risk assessment (PRA) and support National Fire Protection Association (NFPA) 805 license amendment request applications and regulatory reviews.

Researchers will examine the initial impact of the arc to primary equipment and the subsequent damage created by the initiation of an arc in the ZOI (e.g., secondary fires). To meet the goals of this test program, experiments will be conducted to explore the basic configurations and effects of HEAF events. The equipment to be tested in this study primarily consists of switchgears and bussing components.

This project will be operated as part of a larger international OECD/NEA effort. The NRC will be leading the physical testing and instrumentation of equipment at the designated test laboratory. International member countries participating in the project will provide equipment to be tested as well as technical expertise. Figure 4 depicts a 480 V load center undergoing HEAF testing as part of the recent Japanese test program to investigate HEAF events in context of the Onagawa event.

Figure 4. **480 V load center undergoing HEAF Testing: before arcing (left) after arcing (right)**

2.2 Background

As defined by the Institute of Electrical and Electronic Engineers (IEEE), switchgear components are classified as low, medium, and high voltage which corresponds to less than 1 kV, 1 to 100 kV, and greater than 100 kV, respectively. The majority of the testing of these components has focused on exposure to personnel and worker safety. The testing to be conducted in this project is driven by a need to better understand the HEAF phenomenon within an NPP systems framework, the effects on secondary combustibles ZOI, and how to quantify and model these fire risks.

The initial HEAF impacts are important in understanding the structural integrity of the component during overpressure as well as the potential for catastrophic equipment failure. Understanding the heat exposure effects can further define the ZOI. Quantifying this ZOI from a HEAF is particularly important when analyzing the arc effects on secondary combustible materials (e.g., transient combustibles, adjacent equipment, exposed electrical cabling). This provides the basis for subsequent damage, which may result from an ensuing fire. This ZOI approach can create fire scenarios with extensive target sets resulting in high conditional core damage probabilities (CCDP) and little to no solutions for risk mitigation and has been identified as a high priority for future research.

Currently, NUREG/CR-6850, Appendix M discusses the analysis of HEAF and surrounding combustibles as well as the relevant assumptions applied during the analysis. The assumptions were developed from analysis of NPP incidents, previous studies and expert judgment. The majority of the events occurred in 4160 V switchgear/bussing equipment; however, some failures have occurred in 480 V and 6900 V. For the more intense arcing incidents, adjacent cabinets and secondary combustibles (e.g., cables) were impacted by the HEAF. The ZOI for HEAF events is intended to capture the damage generated during the energetic phase. The following is a summary list of the current assumptions from Appendix M of NUREG/CR-6850.

- The initial arcing fault will cause destructive and unrecoverable failure of the faulting device
- The next upstream over-current protection device will trip open
- The release of copper plasma and/or mechanical shock will cause the next directly adjoining/adjacent switchgear and/or load center cubicles within the same bank to fault
- Subsequent fires will burn consistent with a fire intensity and severity consistent with the methodology presented in Appendix G of NUREG/CR-6850
- Unprotected cables that drop into the cabinet will ignite
- Any unprotected cables in the first overhead cable tray will be ignited concurrent with the initial arcing event provided that the tray is within 1.5 m vertical distance of the top of the cabinet
  - Fire will spread to other trays consistent with the treatment of cable tray fires described in the methodology
  - This assumption also applied to trays located 0.3 m in any horizontal direction of the impacted cabinet or duct
  - Cables in fire wrap or conduit are considered protected
- Any vulnerable component within 0.9 m horizontally in front or in the rear of the cabinet will suffer physical damage and functional failure
  - This includes operable structural elements like fire dampers and fire doors, equipment such as cables and transformers, and oil feed lines less than 1” diameter
  - This excludes structural elements such as walls and floors as well as large components and purely mechanical components such as pumps and valves

2.3 Technical Approach

To meet the goals of this test program, experiments will be conducted to explore the basic configurations, failure modes, and effects of HEAF events at the KEMA power test facility in Chalfont, Pennsylvania USA. The equipment to be tested in this study primarily consists of switchgears and bussing components. The HEAF will be initiated inside of the electrical equipment according to IEEE Standard C37.20.7-2007 “Testing Metal-Enclosed Switchgear Rated Up to 38 kV for Internal Arcing Faults. “General characteristics of the arc may be obtained for the different initial voltages (i.e., 480, 4160, and 6900 V). For low voltage equipment, the arc will be initiated by means of a copper wire 2.6 mm in diameter (10 AWG). For medium voltage equipment, a 0.5 mm in diameter (24 AWG) copper wire will be used. (See Figure 5)

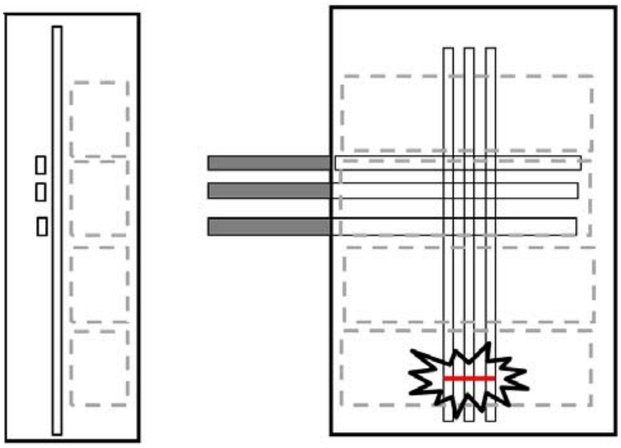


Figure 5. **Initial arc location**

High speed video and instrumentation will be used to best capture relevant information related to the release of energy created during the arcing event. Both active and passive gauges to measure temperature and pressure have been evaluated during an exploratory test program with Sandia National Laboratories [13]. Instrumented cable trays and cables will be placed above the cabinet and evaluated for ZOI damage conditions. It is expected that cables located on cable trays above the location of the HEAF will ignite and support an enduring fire. Where possible, oxygen consumption calorimetry will be used to estimate the heat released from such post-HEAF fire. This testing will use a portable oxygen consumption calorimetry hood developed as part of the Electrical Enclosure HRR testing program by NIST.

2.4 Test Results and Fire Model Improvement

This research program is intended to improve the realism in the current state of the art in fire PRA methodology and will be targeting different types of equipment and different voltage levels to better characterize the ZOI for HEAF events. The data collected will support further classification of risk relevant targets and improve the understanding of HEAF susceptible enclosures. This program is being conducted in collaboration with the OECD.

The current equipment list includes contributions from the Central Research Institute of Electric Power (CRIEPI) Japan, the Japanese Nuclear Energy Safety Organisation (JNES) (now a part of Japan Nuclear Regulator Authority (NRA)), the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Germany, the Korean Institute of Nuclear Safety (KINS), Korea (Republic of) and The Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France. Testing will be performed in 2014 and a final report is expected to be published in 2015.

3. Incipient Fire Detection

3.1 Purpose

Incipient fire detection (IFD) has been used in the telecommunications industries for years to meet prescriptive requirements described in NFPA 76, “Fire Protection of Telecommunication Facilities.” These IFD systems have gained wide acceptance in the telecommunication industry due to the high air change rate in telecommunication facilities which results in high levels of smoke dilution rendering conventional spot type smoke detectors ineffective. Since the mid-1990's the use of these systems have begun their introduction in Canadian and U.S. NPPs as a very early warning fire detection (VEWFD) systems. Air sampling detection (ASD) type smoke detectors are commonly used to meet the Very Early Warning sensitivity and transport time requirements per NFPA 76. These systems have the potential to detect low-energy fires at an early stage, allowing for additional time for operators to response to possible fire threats. However, information regarding the performance of these systems in non-telecommunication type facilities is limited and quantification of the characteristics and benefits of these systems for the application within the PRA has been highly uncertain.

NUREG/CR-6850 provides an empirical fire growth model for electrical cabinet fires that is based, in part, on testing conducted by Sandia National Laboratories, VTT Technical Research Center of Finland, and the empirical "t squared" fire growth model developed by the NIST fire growth model. However these testing programs have provided scarce data on the incipient growth stages of an electrical panel fire and other postulated events or phenomena. This is largely due to the uncertainty and randomness of the incipient phase of fires. Additionally there is a lack of adequate guidance related to PRA treatment to reflect this improvement in plant safety when using these systems. To assist in resolving this issue, the NRC has initiated a confirmatory test program that will analyze the incipient phase of the fire growth stage in electrical panel fires and evaluate both the human behavioral response and detection system performance.

This program includes some consideration for the root cause and fire growth characteristics in cabinets during the incipient fire stage. NRC is also considering the development of a model that would characterize the incipient phase of the fire growth stage based on the information gained during this program.

3.2 Background

ASD systems mechanically draw air samples from the protected area (room or electrical enclosure) and assess these samples for the presence of smoke particles. This allows for the use of filters to remove dirt and dust particles, a common source of false alarms for ordinary nonaspirating smoke detectors [14]. The filtering allows for the detector to have a higher sensitivity than conventional spot detection. These high sensitivity detectors can be used to detect the earliest traces of airborne particles or aerosols released caused by the overheating of materials. As these systems have the potential to provide numerous advantages over conventional systems, there exists the possibility of challenge in adequately implementing this technology to take full effect of the advantage. For example, differences in human interaction, response, system design, installation, maintenance, reliability, and testing need to be taken into consideration to ensure the potential advantages are utilized.

Air aspirated VEWFDS have been used in some U.S. NPPs for over a decade to qualitatively reduce fire risk contributors identified during the Individual Plant Evaluations of External Events (IPEEEs) or for prompt detection to support exemptions. However, only recently has there been an interest in using these systems in the regulatory context to document quantitatively risk reductions per fire PRA methodologies and the NFPA Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition.

In September 2010, the NRC and EPRI issued Supplement 1 to NUREG/CR-6850 / EPRI 1019259 entitled, "Fire Probabilistic Risk Assessment Methods Enhancements" [12]. This report provides interim guidance on questions raised by the pilot plants during their transition to NFPA 805. Section 13 entitled, "Incipient Fire Detection Systems," provides an NRC staff interim position for determining the probability of non-suppression in fire areas that have installed incipient fire detection systems[4]. Because of the lack of information and test data, the interim position limited the applicability of VEWFDS with regard to crediting these systems for fire risk reduction. Several requirements are provided in the FAQ 08-0046, including;

- Smoke detection system shall be an ASD installed as a very early warning fire detection system (VEWFD SYSTEMS) per the requirements of NFPA 76 (2009 version).
  - Per NFPA 76, VEWFD SYSTEMS shall meet two sensitivity criteria;
    - alert threshold of at least 0.2 percent obscuration per foot, and
    - alarm threshold of 1.0 percent obscuration per foot
- ASD shall be installed to monitor component degradation in electrical cabinets
- Cabinet component voltages shall have less than or equal to 250 Volts
- Proper cabinet ventilation must exist to allow ASD VEWFD SYSTEMS to function (systems do not function properly in tightly sealed cabinets)
- No credit should be taken for using ASD VEWFD SYSTEMS to protect rotating equipment or cabinets containing voltage greater than 250 volts.
- The ASD VEWFD SYSTEMS shall be designed and installed by technicians who are trained and qualified to apply NFPA 76 following appropriate vendor guidance, tested in accordance with an appropriate standard including appropriate vendor requirements, and maintained in accordance with manufactures code requirements.
- No credit shall be provided for operator success in identifying ASD VEWFD SYSTEMS alert source and removing power from it during incipient stage.

Based on these recommendations the NRC decided to perform confirmatory testing to evaluate the following aspects of incipient detection systems:

- effectiveness of area wide versus in-cabinet VEWFDS applications, including effects of in-cabinet VEWFDS layout and design on system response.
- comparison between conventional fire-detection systems currently used in NPPs and VEWFDS, including VEWFDS fire signature response to products of combustion from common combustibles found in NPPs.
- human Reliability Analysis (HRA) response and effectiveness of equipment used to locate pre-fire source.
- system reliability
- a scientific evaluation to determine and characterize the root cause and nature of pertinent postulated or observed phenomena and an argument for how VEWFDS may detect and discriminate such phenomena.

[4] *As a matter of clarification, the term incipient fire detection system will not be used in this test plan, instead the term very early warning fire detection systems (VEWFDS) will be used. The use of the term VEWFDS is to reduce any confusion with regard to regulatory applications where licensees have installed conventional non-VEWFDS spot detectors in cabinets or other areas and classified these detectors as incipient detection in licensing documentation.*

- a discussion of the intended or anticipated outcome and benefit of the application of VEWFDS in NPPs.

3.3 Technical Approach

Following a literature review, staff from NIST and NRC performed several site visits to operating NPPs in the United States and Canada, along with visits to non-nuclear facilities. These site visits provided two benefits. First, it was realized early on that the literature review and testing alone would not able to provide answers to all of the program's objectives. With regard to this aspect, the site visits provided information on system availability, reliability and human interaction per plant procedure review and operator interviews. Second, the site visits provided information on the system layout and design being used in plants, which enabled a test plan to be developed that adequately represented the design and use of these systems in plants.

The testing phase included small scale laboratory testing to evaluate system response to various smoke signatures. Researchers selected eleven different materials to represent realistic plant materials and heated at three different heating rates to simulate an incipient fire sources under differing conditions. The full scale testing focused on the impact that room size and ventilation would have on parameters of interest; including in-cabinet detection, area wide detection, and air-return grills. This test program also will evaluate the human response impact for VEWFD's systems since the proposed systems use human intervention as the primary means of mitigating an impending fire.This research program is intended to improve the realism in the current state of the art in fire PRA methodology and provide supplemental guidance for the treatment of VEWFD systems in a PRA model.

References

[1] Electric Power Research Institute and U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, "Fire Probabilistic Risk Assessment Methods Enhancements: Supplement 1 to NUREG/CR-6850 and EPRI 1011989," EPRI 1019259 and NUREG/CR-6850 Supplement 1, Electric Power Research Institute (EPRI), Palo Alto, CA and U.S. Nuclear Regulatory Commission, Washington, DC, 2009.

[2] Babrauskas, V., and R.D. Peacock, "Heat Release Rate: The Single Most Important Variable in Fire Hazard," Fire Safety Journal, 18, pp. 255-272, 1992.

[3] McGrattan, K., et al., "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) – Phase 1: Horizontal Trays," NUREG/CR-7010 Vol. 1, US Nuclear Regulatory Commission, Washington, DC 20555, 2012.

[4] Madrzykowski, D., and D.W. Stroup, "Flammability Hazard of Materials," Section 2, Chapter 3, Fire Protection Handbook, National Fire Protection Association, Quincy, MA, pg. 2-31, 2008.

[5] Hurley, M.J., and R.W. Bukowski, "Fire Hazard Analysis Techniques," Section 3, Chapter 7, Fire Protection Handbook, National Fire Protection Association, Quincy, MA, pg. 3-121, 2008.

[6] Iqbal, N. and M. Salley (2004) Fire Dynamics Tools, NUREG-1805, U.S. Nuclear Regulatory Commission, Washington, DC.

[7] Electric Power Research Institute, Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires, Technical Report-1022993, February 2012 Palo Alto, CA

[8] Chavez, J.M. (1987). An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part I: Cabinet Effects Tests, NUREG/CR-4527/V1, US Nuclear Regulatory Commission, Washington, DC.

[9] Chavez, J.M. and S.P. Nowlen (1988). An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part II: Room Effects Tests, NUREG/CR-4527/V2, US Nuclear Regulatory Commission, Washington, DC.

[10] Nowlen, S.P. (1989) A Summary of Nuclear Power Plant Fire Safety Research at Sandia National Laboratories, 1975-1987, NUREG/CR-5384, SAND89-1359, Sandia National Laboratories, Albuquerque, New Mexico.

[11] Mangs, J., J. Paananen and O. Keski-Rahkonen (2003) “Calorimetric fire experiments on electronic cabinets,” Fire Safety Journal, 38: 165-186.

[12] Electric Power Research Institute and U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” EPRI 1011989 and NUREG/CR-6850, Electric Power Research Institute (EPRI), Palo Alto, CA and U.S. Nuclear Regulatory Commission, Washington, DC, 2005.

[13] Lopez, Carlos, “Evaluation of Heat and Pressure Gauges for High Energy Arcing Fault Tests: A Test Report,” U.S. NRC Project N6981, Sandia National Laboratories (SNL), February 2013.

[14] Custer, Richard L.P.et al. 2002, Design of Detection Systems, SFPE Handbook of Fire Protection Engineering, 3rd Ed., NFPA, Quincy (MA), USA, ISBN: 087765-451-4.

Fundamental investigation on successive fire due to the high energy arcing faults event for the high-voltage switch gears

Koji SHIRAI, Yutaka GODA, Mikimasa IWATA and Tsukasa MIYAGI
Central Research Institute of Electric Power Industry (CRIEPI), Japan

Abstract

High Energy Arcing Faults (HEAF) have the potential to cause extensive damage to the failed electrical components and distribution systems along with adjacent equipment and cables within the zone of influence (hereinafter, ZOI). Furthermore, the significant energy released during HEAF event can act as an ignition source to other components within the area of the HEAF. Risk Assessment Technical Committee in Atomic Energy Society of Japan Committee is currently developing Standard for Internal Fire Probabilistic Risk Assessment issued in the mid of 2014 [1]. In this standard, although the ZOI defined in the NUREG/CR-6850, Appendix M (2005) is tentatively referred [2], the accumulation of experiment data of HEAF for the estimation of the severity factors is highly recommended. In order to understand the critical condition for the successive fire occurrence of the electrical cabinets due to HEAF, several series of 3-phase internal arc tests with two type of high voltage switch gears were executed. This paper presents the fundamental test results and the fire occurrence condition from the released arc energy point of view was investigated. Moreover, according to the test results, numerical tool using the impact analysis code AUTODYN to estimate the pressure increase was investigated.

1. Introduction

The Atomic Energy Society of Japan (AESJ) has been working on development of the “Implementation Standard for Internal Fire Probabilistic Risk Assessment (IFPRA) of Nuclear Power Plants (NPPs)” since fiscal year 2012 [2]. This IFPRA standard prescribes the requirements and specific methods to implement Level 1 PRA for accidents initiated by internal fire at NPPs during power operation. This standard mainly focuses on thermal effects, and excludes other effects, such as component failure due to micro-particle of smoke and mechanical damage due to explosion (HEAF or hydrogen explosion), which could induce large uncertainty on FPRA results because of insufficient scientific knowledge for them.

On the other hand, large electrical discharges, referred to as HEAF, have occurred in NPPs switching components throughout the world [3]. 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). According to the OECD report [4], 11.5% of the fire events collected in the OECD database [5] were relevant to HEAF fire events.

Therefore, it is very important to obtain scientific fire data on the HEAF phenomenon known to occur in NPP through experiments and improve the state of knowledge, develop a mechanistic model to account for the failure modes and provide better characterization of HEAF for the fire prevention countermeasures. The objective of our experimental studies is to clarify the initial impact of the arc to primary equipment and the subsequent damage created by the initiation of an arc (e.g., secondary fires) on secondary combustibles (e.g., electrical control cables) and to investigate the numerical approach to estimate the damage of the electrical cabinets.

2. Experimental Approach

In order to resolve the uncertainty associated with the energetic arcing fault and subsequent fires and quantify the fire effects at common voltage levels, the HEAF experiments using high voltage switchgears were conducted at the High Power Testing Laboratory of the CRIEPI by the request of the Federation of Electric Power Companies of Japan.

The HEAF test program consisted two phases as follows. The equipment considered in our study consists of non-seismic-proof and seismic-proof switchgears.

- Phase-I : Ten HEAF tests using Non-Seismic, Non-Arc-Proof 6.9kV Switchgears
- Phase-II : Three HEAF tests using Seismic, Non-Arc-Proof 8.0kV Switchgears

2.1 Test facility

The Electric Power Engineering Research Laboratory of the CRIPEI, located in the south of about 65km from the centre 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 was newly established in 2001, and laboratory accreditation was granted by the Japan Accreditation Board for Conformity Assessment (JAB) in compliance with ISO/IEC17025. 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 switchgear and gas insulated switchgear can be conducted with the test capacity of current up to 60kA and duration up to 2 seconds.

2.2 Test matrix

The total number of HEAF tests was thirteen, varying the type of switchgears, rating voltage, current and the arc discharge locations. Test matrix was setup referring “JEM1425-2011, Appendix A -Internal Fault-” [6] as shown in Table 1.

For the HEAF tests, eight 6.9kV non-seismic/non-arc-proof switchgears and two 8.0kV seismic/non-arc-proof switchgears were tested in the unloaded condition (no primary load attached to them). Arcing current was set to 20kA considering maximum 3 phase short-circuit current from the designated power system. Referring the standard of JEM1425-2011, arc duration was set to the range between 0.1 and 1.0 sec. Moreover, from the safety point of view, the longer arc duration 2.0 sec was also considered. As arc discharge points, secondary bus in the cable room and VCB terminal were selected and the arc was initiated by means of a copper wire 0.5 mm in diameter. The cabinet doors were closed to represent events that may occur under normal operation.

The test switchgears 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 ignition.

2.3 Test equipment

In phase-I program, the total eight units of non-seismic-proof and non-arc-proof “3-6.9kV metal-clad switchgears” were prepared as shown in Figure 1 and four series of the HEAF tests were executed. In test series 1, series 2, and series 3, cabinets A+B, cabinets C+D and cabinets E+F+G+H were used, respectively. In series 4, cabinets A+B were reused by reconditioning after test series 1.

In phase-II program, the total two units of seismic-proof and non-arc-proof “3-8.0kV metal-clad switchgears” were prepared as shown in Figure 2 and test series 5 were executed using cabinets I+J.

2.4 Measurement

During the tests, arc intensity (e.g., arc power, rated current and voltage), arc duration, inner pressure inside the cabinet by pressure sensors, surface temperature of the cabinets by thermography and the passive temperature within ZOI were measured. Additional instrumentation included the high-speed video cameras. Moreover, in case of series5, the instrumented cable trays were placed in the vicinity of switchgear to investigate the thermal damage or the potential fire resulting from the arc event.

Table 1. **HEAF test matrix and test results**

Test Case	**Arc Discharge Location**			**Voltage** (kV)	**Current*** (kA)	**Duration** (sec)	**Fire**
	Cabinet	**Room**	**Location**				
Phase-I : Use of eight Non-Seismic-proof, Non-Arc-Proof Switchgears							
1-1	A	Upper	Secondary bus	6.9	18.9	0.1	No
1-2	B	Upper	Secondary bus			0.3	No
2-1	C	Upper	Secondary bus			0.5	No
2-2	D	Upper	VCB Terminal			0.5	No
3-1	E	Upper	Secondary bus			1.0	No
3-2	F	Upper	VCB Terminal			1.0	No
3-3	E	Lower	Secondary bus			1.0	Yes**
3-4	F	Lower	VCB Terminal			2.0	Yes***
4-1	A	Lower	VCB Terminal			2.0	Yes***
4-2	B	Lower	VCB Terminal			1.0	No
Phase II : Use of two Seismic-proof, Non-Arc-Proof Switchgears							
5-1	I	Upper	Secondary bus	8.0	40.0	0.2	No
5-2	I	Lower	VCB Terminal			0.2	No
5-3	J	Lower	VCB Terminal			0.5	No

Remarks:
* Arcing current: Max. 3 phase short-circuit current from the designated power system
** Self-extinction observed after 20min.
*** Extinction Work executed by Handy Water Spray at 7-10min after ignition.

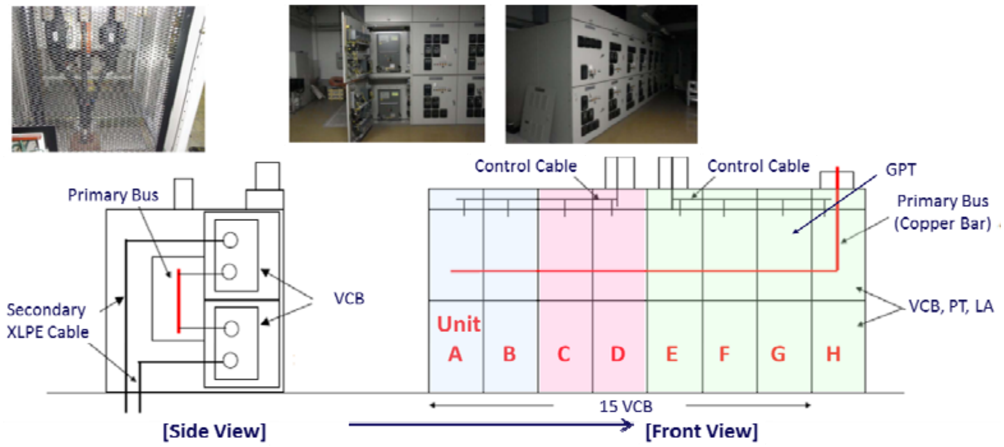


Figure 1. **Non-seismic and non-arc-proof "3-6.9kV metal-clad switch gears" for Phase-I**

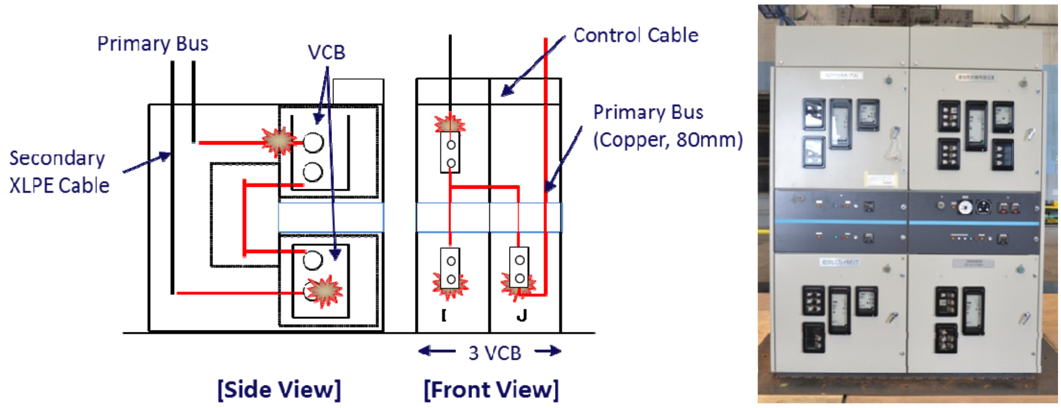


Figure 2. **Seismic-proof and non-arc-proof "3-8.0kV metal-clad switchgears" for Phase-II**

3. HEAF Test Results

3.1 Damage to components

Figure 3 shows the damage of the components observed in the Phase-1. In case that the arc was ignited in the upper cable room or at the VCB terminal located in upper layer with duration 1.0 second, there was no remarkable damage of the control cables in the upper duct although highly pressurized hot gas was released from deformed roof and side panels. On the other hand, in case that the arc was ignited at the VCB terminal located in lower layer with duration 2.0 seconds, the door opening due to the high pressure and the melting of the front panel of VCB were observed. After 15 minutes, fire was actively extinguished by handy water spray according to the safety procedure of the test facilities.

Figure 4 shows the test equipment layouts and test images for test series 5. Figure 5 shows the damage of the components observed in the test 5-1. In case that the arc was ignited in the upper cable room with duration 0.2 second, the roof and rear panels were came off and the cable tray were deformed remarkably due to the impact of the came-off roof panel.

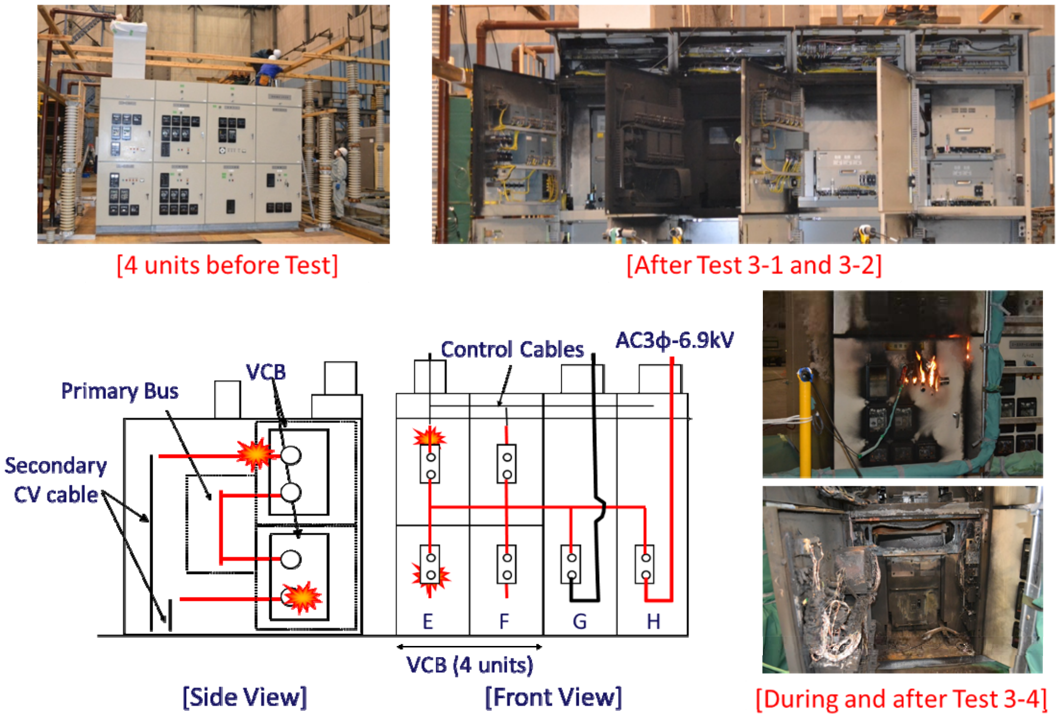


Figure 3. **Typical damage of equipment observed in Phase 1**

(Test 5-1: Upper/ Secondary bus) (Test 5-3: Lower/ VCB Terminal)

Figure 4. **Test equipment layouts and test images for test series 5**

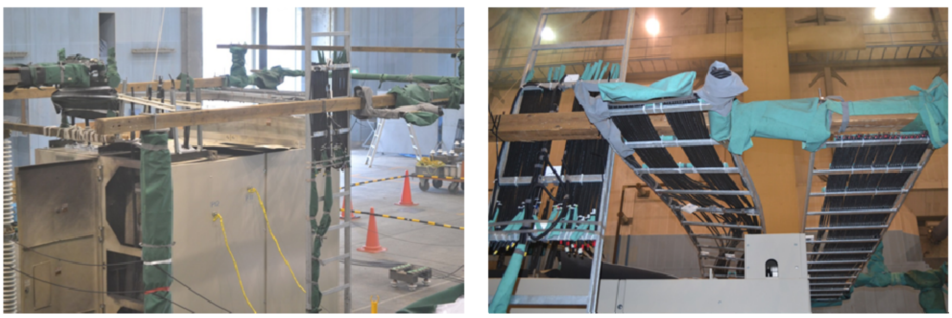
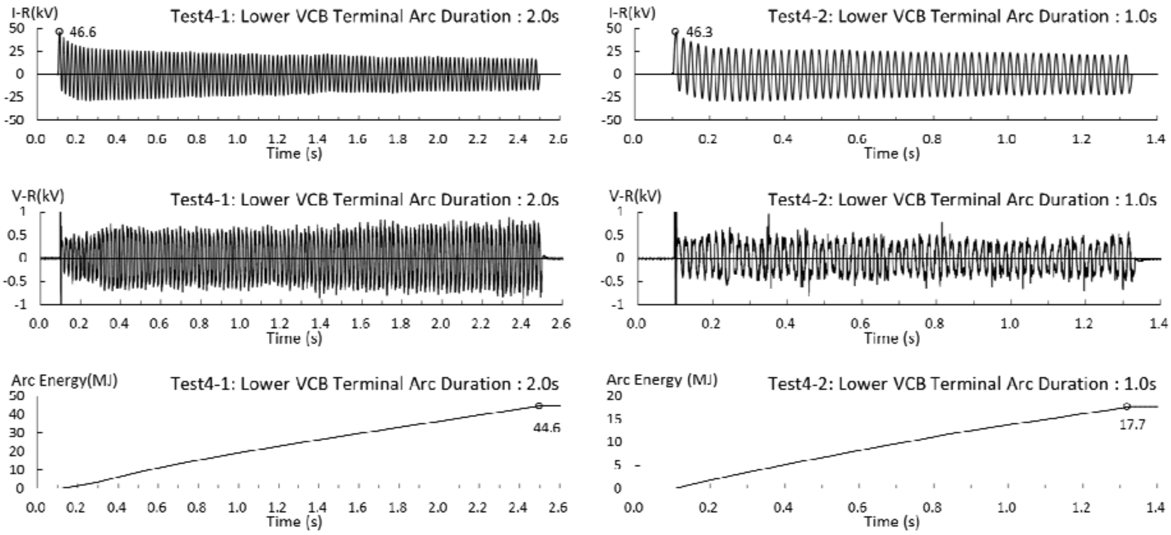


Figure 5. **Damage of the equipment observed in Test 5-1**

3.2 Arcing energy

Figure 6 shows the example of the measured arc current, voltage and energy. For example, in the test 4-1, the average arc current was 14.9kA and the arc voltage varied from 500 to 800V. Figure 7 shows the relationship between arc duration and measured arc energy. Arc energy in case of discharge in the cable room was 20% larger than that in case of discharge at the VCB terminal. In both cases, below the total arc discharge energy 25MJ (below arc duration 1.0s), as highly pressurized hot gas was released from detached roof and side panels, there was no remarkable damage of the control cables in the duct and within the zone of influence, and the successive fire was not detected. On the other hand, over the total arc discharge energy 25MJ (longer arc duration 2.0 sec), the fire occurrence was detected and the fire extinction activities to remove the toxic gas release were needed. From these test results, it seems that the prevention of the successive fire or the mitigation of the influence for the equipment within ZOI may be accomplished if the arc duration or release energy is properly controlled below 1.0s.



(Test 4-1: Lower/ VCB Terminal) (Test 4-2: Lower/ VCB Terminal)

Figure 6. **Example of the measured arcing current, voltage and energy**

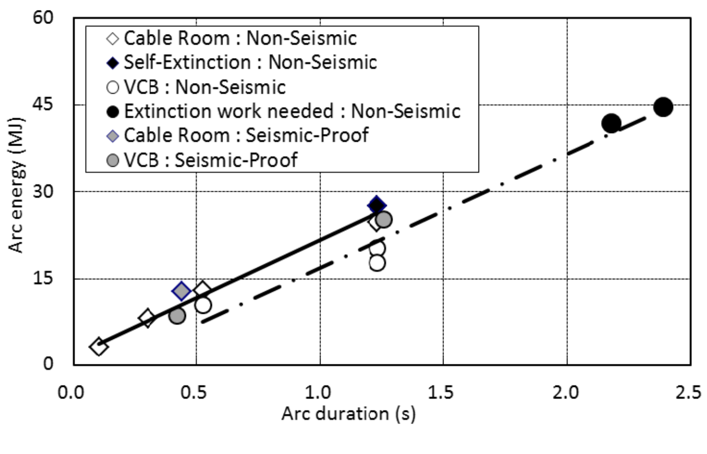


Figure 7. **Relationship between arc duration and measured arc energy**

3.3 Inner arc pressure

Neglecting all transient and hydrodynamic effects, the discharge of an arc in a cabinet can be treated as an ideal gas within constant volume system. If an amount of energy ΔQ is injected into the volume, the change in pressure Δp is expressed as follows [7].

$$\Delta p = (\gamma - 1) \cdot \frac{k_p \Delta Q}{V}$$

where volume V, adiabatic coefficient γ=1.4, which is often used in high voltage switchgear application [6]. Thermal transfer coefficient k_p is the fraction of energy going into raising the gas pressure, while for copper bus bar k_p= 0.53 [7]. In case of test 1-1, at an arc duration 0.1s, an arc energy 3MJ was found. Therefore, at a volume of $4m^3$, the above theoretical pressure of 159kPa was obtained.

Figure 8 and Figure 9 show the relationship between arc duration and measured inner pressure and pressure time histories. Although the structure of the seismic-proof cabinet was strengthened to bear the severe seismic force, it could bear pressure at most up to 70kPa. As only one peak was appeared in any time history, it seems that the panels (roof or side or rear) were highly deformed or opened or partially detached just after the first arrival of the shock wave due to the arc discharge. Moreover, as the several pieces of the broken bolts were scattered at least 15m away from cabinet, they should be considered as a projectile which might induce the local damage to the surrounding equipment.

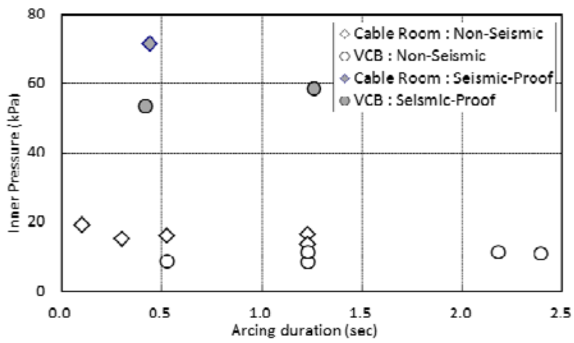


Figure 8. **Relationship between arc duration and measured inner pressure**

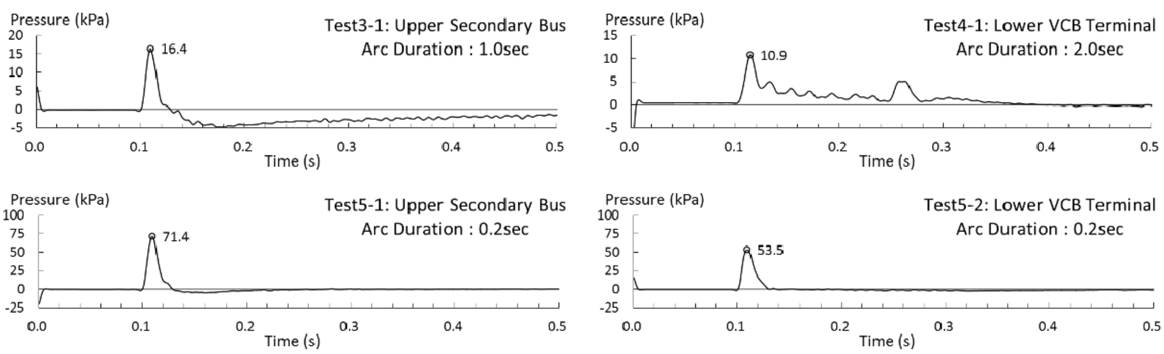


Figure 9. **Example of time histories of measured inner pressure**

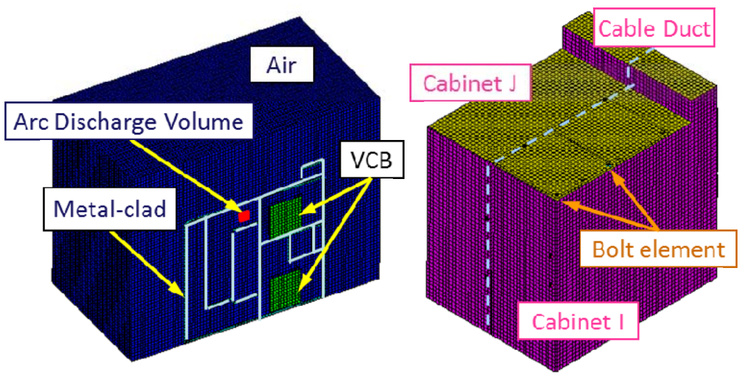
4. Numerical Approach

4.1 Numerical model

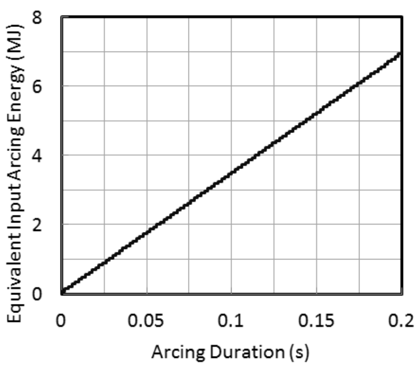
To determine the zone of influence due to the HEAF event, it is very essential to estimate the structural weakest point of the cabinets and the thermal propagation of the hot gas through the damaged cabinets to the environment. Therefore, in order to investigate the accuracy of the numerical approach for the realistic pressure rise due to AC arc ignition in the cabinet, impact analysis code "AUTODYN (Ver.13)" was applied considering the interaction between arc pressure and panel deformation [8].

Figure 10 shows the numerical model for test case 5-1 with two-full scale electric panels (Unit I & J, Seismic-proof / non-arc-proof type) with upper cable duct. The in-house subroutine was developed and installed to AUTODYN to represent designated arc energy by pulse function accumulation. In this case, the energy increment was set to 69kJ and charged every 2ms in the designated arc discharge volume in the cabinet. Moreover, the rupture of clamping bolts on the top and side panels was considered by use of beam rupture element, in which the threshold value for the rupture was set to 420MPa of the equivalent plastic stress.

Table 2 shows the analysis condition. Thermal transfer coefficient k_p described above was set to 0.53 considering the material type (copper) of bus bars [9].



(Three-Dimensional Numerical Mesh)



(Equivalent arc energy input)

Figure 10. **Numerical model for test case 5-1 (Unit I & J, Seismic-proof / non-arc-proof type)**

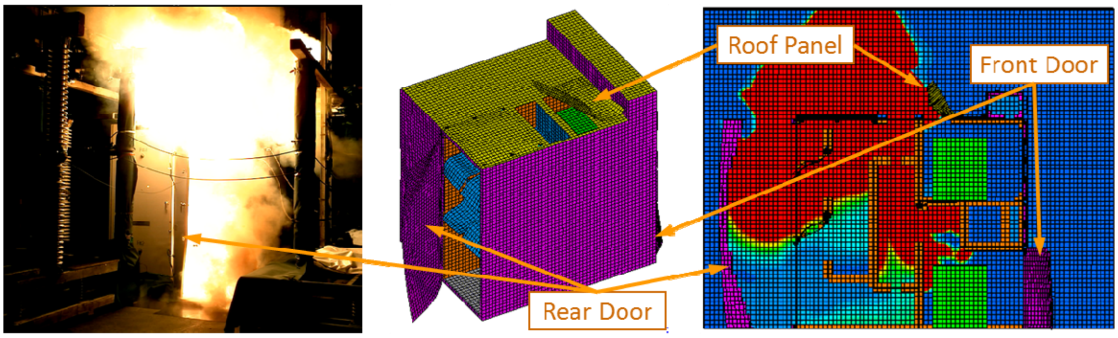
4.2 Application to HEAF tests

Figure 11 and Figure12 show the deformation of the cabinet, temperature distribution and the pressure profile obtained from numerical analysis. Although calculated peak pressure value 65kPa at the upper area of the rear door of the cabinet I was slightly less than the measured value 71kPa, the calculated response period of the pressure seems to be in a good agreement with the measured one. Moreover, as the detachment of the roof panel and the opening of the rear door due to the collision of the tightening bolts were well represented, it is found that the structural weakest point, pressure increase and heat release can be interpreted by the proposed numerical analysis methodology. However, it should be noted that huge elapsed time (1 week for 0.05s) is necessary to accomplish total arc duration with standard PC system.

Table 2. **Analysis condition for HEAF Test with AUTODYN**

Test Case	No.5-1
Cabinet Type	Seismic-proof / Non-arc-proof
Main Panel Thickness	3.2mm
Arc Current and Voltage	20.0kA x 8.0kV
Arcing Duration	0.2s
Arc Discharge Location	Secondary bus in cabinet I
Arc Discharge Volume	1000 cm^3
Arc Energy (Thermal transfer coefficient)	12.8MJ (k_p=0.53)
Material of Electrode	Copper (Cu)
Gap between Electrodes	50mm

Remarks
– Lagrange solver : Metal-clad Plates
– Euler solver : Inner and Circumferential Air
– Fill solver : VCB (rigid)
– Clamping bolts on the top and side panels : Beam rupture element
– Energy increment (69kJ) was charged every 2ms in the arc discharge volume



(HEAF Test View) (Deformation) (Temperature Distribution)

Figure 11. **Deformation and temperature distribution (Test case 5-1, at 0.04s after ignition)**

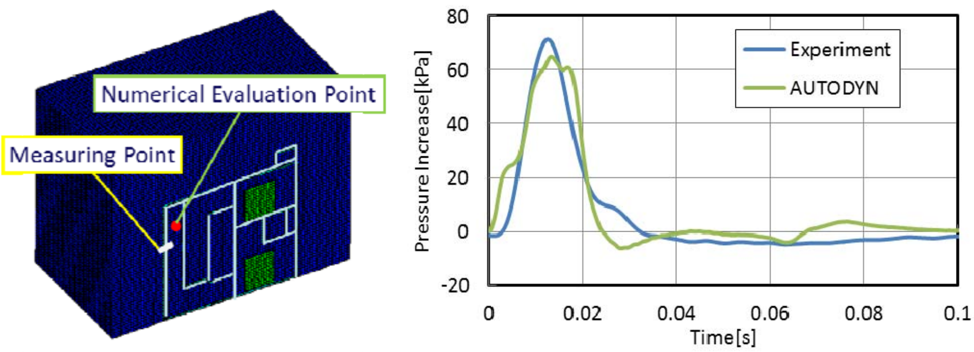


Figure 12. **Pressure profile inside cabinet I (Test case 5-1)**

5. Conclusion

In order to clarify the mechanism of pressure rise in the cabinet and thermal propagation through cabinets due to arcing fault fire and evaluate the zone of influence for the adjacent cabinets and surrounding equipment, thirteen HEAF tests (arcing duration from 0.1 to 2.0s) using eight non-seismic and two seismic-proof / non-arc-proof cabinets, at High Power Testing Facility in CRIEPI (Yokosuka, Japan) had been executed. As a result, below the total arc discharge energy 25MJ (below arcing duration 1.0 sec), as highly pressurized hot gas was released from detached roof and side panels, there was no remarkable damage of the control cables in the duct and within the zone of influence, and the successive fire was not observed. On the other hand, over the total arc discharge energy 25MJ (longer arcing duration 2.0s), the fire occurrence was detected and the fire extinction activities to remove the toxic gas release were needed. Moreover, according to the test results, applicability of the numerical tool using the impact analysis code AUTODYN to estimate the pressure increase was well-benchmarked.

References

[1] Development of the Implementation Standard for Internal Fire Probabilistic Risk Assessment of Nuclear Power Plants, PSA 2013 – International Topical Meeting on Probabilistic Safety Assessment and Analysis, Sept. 22-27, 2013

[2] EPRI/NRC (2005), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities", NUREG/CR-6850, 2005

[3] H.P.Berg1and M. Röwekamp (2011), "Chap.7, Investigation of High Energy Arcing Fault Events in Nuclear Power Plants", Chap.7 Nuclear Power -Operation, Safety and Environment, ISBN 978-953-307-507-5, INTECH, July, 2011 http://www.intechopen.com/articles/show/title/investigation-of-high-energy-arcing-fault-events-in-nuclear-power-plants

[4] Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) (2013), "OECD FIRE Project - TOPICAL REPORT No. 1, Analysis of High Energy Arcing Fault (HEAF) Fire Events", NEA/CSNI/R, 25.June, 2013 http://www.oecd-nea.org/documents/2013/sin/csni-r2013-6.pdf

[5] OECD/NEA, Committee on the Safety of Nuclear Installations (CSNI), "OECD FIRE Database, Version: OECD FIRE DB 2012:1", Paris, France, December 2012

[6] Japan Electrical Manufacturers' Association (JEMA) (2012), "A.C. metal-enclosed switchgear and control-gear for rated voltages above 1kV and up to and including 36kV", Standards of the JEMA.

[7] Babrauskas, V. (2010), "Electric Arc Explosions", Proc. 12th Intl. Conf. Interflam, pp. 1283-1296, 2010

[8] Shirai, K. et al., (2013), "Experimental Studies on Electric Arc Explosion Events using the 6.9kV Switchgears subjected to High Energy Arcing Fault", 22nd International Conference on Structural Mechanics in Reactor Technology (SMiRT22), 13th International Post-Conference Seminar on Fire Safety in Nuclear Power Plants and Installations, Sep., 2013

[9] Iwata, M. et al., (2010), "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

EXPERT JUDGMENT: AN APPLICATION IN FIRE-INDUCED CIRCUIT ANALYSIS

Gabriel Taylor P.E., Nicholas Melly, Tammie Pennywell
U.S. Nuclear Regulatory Commission[1], USA

Abstract

Expert judgment has been used to develop both qualitative guidance and quantitative estimate of hot short-induced spurious operation of equipment as a result of fire damage to electrical cables located in a nuclear power plant (NPP). This paper summarizes the research efforts and explores the quantitative risk differences. The U.S. Nuclear Regulatory Commission (NRC) sponsored Brookhaven National Laboratory to facilitate this project with the Electric Power Research Institute (EPRI) under the NRC-RES/EPRI Memorandum of Understanding. The first phase of using expert judgment followed a Phenomena Identification and Ranking Table (PIRT) process using a group of electrical engineering experts to identify and rank the parameters that influence fire-induced cable damage. In addition to the PIRT results, this electrical engineering expert panel also provided technical consensus on several long time contentious fire safety issues related to electrical circuit analyses. Specifically, the electrical engineering expert panel results categorized circuit configurations as being possible, implausible or incredible; which provided strong technical recommendations to support regulatory activities. The results of the PIRT panel also identified areas where knowledge is low and suggested future research prioritized on risk insights. A second expert panel of fire PRA experts was convened with the goal of using expert judgment for quantifying probabilities and durations of hot short-induced spurious operations caused by fire damage to electrical cables, utilizing the Senior Seismic Hazard Analysis Committee (SSHAC) process. Following this process, this second panel results are expected to represent the opinion of the relevant informed technical community. These results are mainly derived for performance-based applications. The updated results of this expert judgment exercise are used in a few select case studies to evaluate the change in conditional risk from the original 2005 fire PRA methodology.

1. Introduction

A vital component for safe nuclear power plant (NPP) operation is the electrical cables. Operating experience and testing has shown that fire-induced cable failures can adversely affect operators' ability to safely shutdown the reactor. Recently, two expert panels were formed to advance the state-of-the-art for modeling fire-induced cable failures in a fire PRA. The project was conducted jointly between the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES) (NRC-RES) and the Electric Power Research Institute (EPRI) under their Memorandum of Understanding (MOU) for collaborative research. The experts on both panels were made up of equal numbers regulator (NRC staff and NRC contractors) and EPRI representatives (nuclear utility staff and contractors). The first panel focused on identifying how various electrical circuit and cable

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

characteristics influence fire-induced cable failure, while the second panel used the results to develop best-estimate conditional probabilities and durations for various fire-induced cable failure modes and configurations. Brookhaven National Laboratory (BNL) was responsible for facilitating the meetings and documenting the conclusions of the first panel, defining the technical approach used for evaluating the probabilities, assessing the probability & duration distributions and documenting the conclusions for the second panel. NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)" [1] documents this work and the results. The results of both panels' work is now being used to update the current state-of-the-art method for conducting fire PRAs, documented in EPRI TR 1011989 and NUREG/CR-6850 [2], as well as updating applicable regulatory guidance.

1.1 Description of the fundamental safety issue

Electrical cables provide the path for transmitting electrical energy (e.g., power, control, instrumentation signals) between two points in an electrical circuit, while simultaneously maintaining the electrical integrity of the signals from each other and from other media that can cause interference. As learned from the 1975 fire at the Browns Ferry NPP and from numerous fire testing programs, the effects of fire on electrical cable can have varying impacts on electrical cable functionality ranging from the loss of system control to spurious operations of systems and components. From a safety perspective, a fire affecting the ability of an electrical cable to perform its function can compromise the operators' ability to safely control and shutdown the plant. The quantity of electrical cable within a NPP varies from a few hundred miles to nearly 1,000 miles [3]. Given the large quantity and types of electrical cable in NPPs and the fact that cables represent a large fraction of the total combustible loading, the importance of protecting them, especially those associated with safety systems, from the adverse effects of fire is necessary. Deterministic and performance-based approaches to fire protection programs are currently available for use in the regulatory environment. In the deterministic approach, strict levels of performance and circuit protection provide a minimum level of safety while the fire PRA methods relies on a rigorous process to locate the cables, model and calculate the risk associated with the effects to electrical cables from fire-induced damage.

1.2 Deterministic fire protection requirements related to circuit analysis

The Browns Ferry Nuclear Power Plant fire of 1975, illustrated the safety significance that fire can pose to NPP operations. In response to the near miss accident, the NRC developed deterministic fire protection requirements and guidance[2]. The objective of the fire protection requirements and guidance is to provide reasonable assurance that one train of systems necessary to achieve and maintain hot shutdown is free of fire damage. This includes protecting circuits whose fire-induced failure could prevent the operation, or cause maloperation, of equipment necessary to achieve and maintain post-fire safe shutdown. This protection is the third echelon of a fire protection defense-in-depth philosophy where the first two echelons include preventing fire from starting, and rapidly detection and suppressing any fire that do occur.

In an effort to make nuclear power plants safe in light of the Browns Ferry Fire, the NRC used the best data available at the time (late 1970's) coupled with engineering judgment to develop explicit prescriptive requirements which plants in various phases of operation, construction, and development were required to follow per a regulatory backfit. Because of the time constraints, perceived risk significance, and lack of understanding the fire induced failure mechanisms for electrical cable, the guidance has been perceived to be either over conservative, unclear, or impractical for plants to

---

[2] *Fire protection regulations and guidance refers to Title 10 of the Code of Federal Regulations, Section 50.48 (10 CFR 50.48), 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 3, Appendix R, Section 9.5-1 of the Standard Review Plan (SRP), NUREG -0800, and the licensees individual fire protection licensing bases.*

implement. These complications resulted in numerous generic communications to help clarify the requirements and issuance of exemptions from specific requirements which could not be met on a plant by plant basis. In the late 1990's the NRC stopped temporarily suspended inspecting fire-induced circuit failure until additional research could be conducted to better understand fire-induced failure modes, and better implementation guidance for fire protection circuit analysis and safe shutdown analysis be issued.

1.3 Fire PRA method

In 2004, the NRC amended is regulations to allow for a risk informed performance-based approach to fire protection to incorporate by reference the 2001 edition of the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" 2001 edition. To support applicant's use of the NFPA 805 standard, the NRC and EPRI developed a state of the art method for developing a fire PRA as documented in EPRI 1011989 (NUREG/CR-6850), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." This report presents a multi-task, iterative process for quantifying fire risk. Cable and circuit analyses are described in Task 3, "Fire PRA Cable Selection;" Task 9, "Detailed Circuit Failure Analysis;" and Task 10, "Circuit Failure Mode and Likelihood Analysis." Task 3 focuses on identifying and locating the electrical cables associated with the fire-PRA components identified in Task 2, "Fire PRA Component Selection." Task 9 provides a method to perform a deterministic circuit analysis aimed at identifying the possible circuit failure modes and excluding any cables that cannot have an adverse effect on the PRA success criteria. Task 10 assigns a probability of spurious operation (conditional on the occurrence of fire) based on the deterministic circuit analysis (Task 9) and test results performed by the Nuclear Energy Institute (NEI)/EPRI in the early 2000s.

1.4 Fire-induced circuit failure test data

Fire tests focusing on nuclear safety have been conducted since the early 1970s. The majority of the early fire testing focused on quantifying the fire phenomena (e.g., heat-release rate, heat flux, flame spread, etc.) of potential NPP fire scenarios. In 2001, NEI in collaboration with EPRI conducted 18 fire tests which evaluated the electrical performance of cables under severe thermal fire conditions. At the time, there was varying opinion on the likelihood of fire-induced hot shorts[3] that can cause equipment to spuriously operate. The results of this testing is documented in EPRI TR 1003326 [4] and demonstrated that hot short-induced spurious operations [4]of components associated with cables damaged by fire occur more frequently than had previously been assumed. Following these tests, the NRC conducted a facilitated workshop and EPRI conducted an expert elicitation to quantify the likelihood of fire-induced hot shorts based on these test results. The results of this effort are documented in EPRI TR 1006961 [5] and concluded that the likelihood of fire-induced hot shorts to be in the range of 0.005 – 0.75 depending on the cable configuration. The likelihood estimates developed by EPRI were subsequently incorporated into EPRI 1011989 (NUREG/CR-6850) under Task 10. In Task 10, Tables 10-1 through 10-5 provide best estimates based on various cable characteristics, such as insulation type, raceway type, hot-short failure mode (intra-cable or inter-cable), armoring, and whether a control power transformer was used as the circuits' power source.

Given the limited number of tests performed by NEI/EPRI in the early 2000s and to address several regulatory issues documented in Regulatory Issue Summary (RIS) 2004-03, "Risk-Informed

[3] *Individual insulated conductors of the same or different cables that come in contact with each other and that may result in an impressed voltage or current on the circuit being analyzed (per PIRT panel's definition in NUREG/CR-7150, Volume 1 [1]).*

[4] *A circuit fault mode wherein an operational mode of the circuit is initiated (in full or in part) due to failure(s) in one or more components (including cables) of the circuit. For example, a pump (starting or stopping) or a valve spuriously repositioned (per PIRT panel's definition in NUREG/CR-7150, Volume 1 [1]).*

Approach for Post-Fire Safe-Shutdown Associated Circuits Inspections, [6]" NRC-RES sponsored two subsequent fire test projects focused on collecting data on a variety of circuit characteristics for alternating current (AC) and direct current (DC) systems [7, 8]. The NRC data obtained was also used to further develop a fire model to predict cable damage.

2. Expert Panels

2.1 Objectives of the panels

One of the objectives of the panels was to advance the state-of-the-art methods in quantifying the risk of fire-induced circuit failures beyond that presented in EPRI 1011989 (NUREG/CR-6850). Due to scarce (or in some instances non-existing) test data and/or analyses, it was decided to use a structured expert judgment process to derive positions to represent the knowledge of the broad scientific community. Because of the technical complexity of the problem, and the different skills needed, the expert judgment process was divided into two panels. The first panel involved a group of electrical engineering experts who performed a Phenomena Identification and Ranking Table (PIRT) exercise. The PIRT panel's specific objectives included: 1) identifying the phenomena that lead to fire-induced hot shorts causing the spurious operation of equipment important to safety; 2) ranking the influencing parameters affecting fire-induced hot shorts, and assessing the current level of knowledge for each of the identified phenomena; and 3) providing consensus technical positions on long-standing fire-protection circuit issues. Thus, the experts on the PIRT panel provided the technical basis for why and how electrical cables and circuits fail from fire effects. This PIRT study is based on established expert elicitation methods employed by other PIRT panels that the NRC used in other technical areas where tests or analyses alone could not address the technical issues with the desired level of certainty [9, 10].

The second panel consisted of experts knowledgeable in fire PRA. The specific objective of the PRA expert elicitation panel was to develop best-estimates for the probability and duration of spurious operation due to fire-induced cable damage using the information derived from the PIRT panel, the complete set of results from the tests listed above, and their own expertise. The Senior Seismic Hazard Analysis Committee (SSHAC) process was used by this panel, so the results obtained are expected to represent the opinion of the relevant informed technical community.

2.2 PIRT process and results

A complete history of the PIRT results are presented in NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure"[1]. This section provides a summary of the results obtained.

To complete the PIRT process, two Figures of Merit were defined, namely:

Spurious Operation[5]: After fire-induced cable damage has occurred to an appropriate conductor in an electrical circuit resulting in a hot short(s), a spurious operation(s) of the component occurs driven by the same electrical circuit.

Duration of Spurious Operation: Duration is the amount of time during which the fire-induced hot short transfers voltage or current to an appropriate conductor of a specific component or device that then can cause the component to move or travel in the undesired direction.

[5] *Spurious operation is defined as a circuit fault mode wherein an operational mode of the circuit is initiated (in full or in part) due to failure(s) in one or more of the circuit's components (including cables). For example, such modes include a pump (starting or stopping) or a valve spuriously repositioning.*

Based on detailed discussions of the various aspects of hot short-induced spurious operation, the PIRT panel identified 16 influencing parameters:

Conductor Count	Cable Grounding Configuration (AC only)
Fire Exposure Conditions	Power Supply Voltage (AC only)
Cable Routing / Raceway Type	Armoring: Grounded vs. Ungrounded (AC);
Cable Raceway Fill	Armored vs. Unarmored Cable (DC)
Conductor Insulation Type	Cable Wiring Configuration
Cable Aging	Conductor Size
Cable Jacket Insulation Material	Fire Suppression
Time-Current Circuit Characteristics	Circuit Latching
	Grounded vs Ungrounded Circuit (AC only)

The PIRT panel recommended additional analyses of test data which resulted in the development of NUREG-2128, "Electrical Cable Test Results and Analysis During Fire Exposure (ELECTRA-FIRE), A Consolidation of Three Major Fire-Induced Circuit and Cable Failure Experiments Performed Between 2001 and 2011" [11]. Using the results of this document allowed for the experts on the PIRT panel to make further refinements and rank the influencing parameters based on importance for both spurious operation and duration.

For the first Figure of Merit, spurious operation, the PIRT panel indicated the following parameters as having a HIGH IMPACT on likelihood:

- Wiring Configuration
- Conductor Insulation Material (for inter-cable shorts)
- Grounding Configuration
- Cable Raceway Routing Configuration (panel wiring)
- Raceway Fill (bundle configurations)

For the second Figure of Merit, duration, the PIRT panel identified the following parameters as having a HIGH IMPACT:

- Fire Exposure Conditions
- Time-Current Characteristics
- Wiring Configuration
- Cable Raceway Routing (Panel Wiring)
- Cable Raceway Fill (bundles)

In addition to the ranking of parameter importance, NUREG-2128-(ELECTRA-FIRE) also allowed for the PIRT panel members to elaborate on a newly identified (but previously postulated in NUREG/CR 6834[13]) failure mode wherein multiple shorts to ground cause a DC circuit to spuriously operate. This failure mode is referred to as "ground fault equivalent hot shorts," abbreviated GFEHS and is the only inter-cable failure mode observed in DC circuit testing. Illustrative examples of GFEHS are shown in Figure 1. This failure mode can impact circuits routed in dedicated conduits since this particular phenomenon occurred between cables co-located on the same raceway, as well as between cables located on different raceways provided they both belonged to the same ungrounded common-power supply. The results from the data analysis of NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results", shows that these events occurred in nearly every test during the intermediate-scale testing.

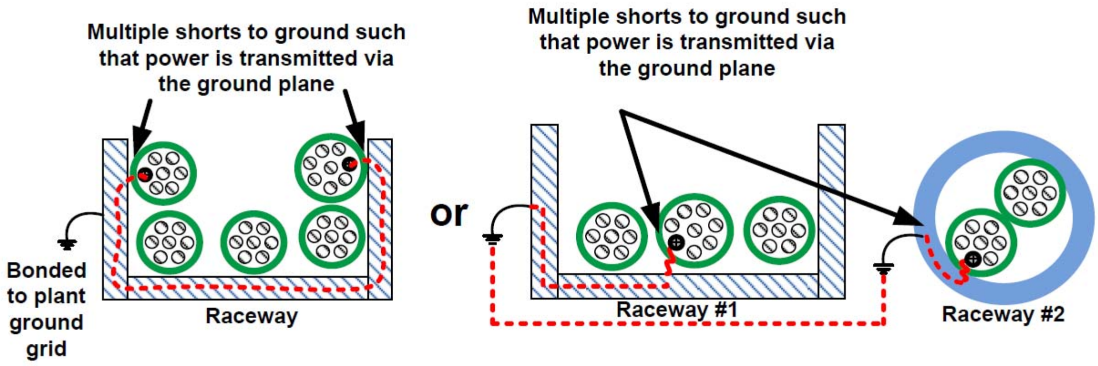


Figure 1. **Illustrative Examples of Ground Fault Equivalent Hot Shorts**

Inter-cable ground fault equivalent hot short, is a complex failure mechanism that was observed in DC testing. This case is only applicable to circuits powered by an ungrounded source that might include an ungrounded DC battery bank, an ungrounded AC CPT, or an ungrounded AC-power distribution source. This case postulates that one leg of the power source becomes grounded due to conductor shorting resulting from fire-induced failures, allowing power transmission via the common ground plane to another conductor that also is grounded. The ground plane may be available via grounded shield-wraps, grounded drain wires, cable armoring, metal raceways (e.g., trays, conduits), or grounded conductors within a cable (e.g., grounded spare conductors). All grounds are assumed to be associated with the same plant-wide ground plane.

Once the formal PIRT process was complete, the electrical expert panel continued to develop an advanced understanding of fire-induced cable failure by developing tables for the follow-on PRA expert elicitation panel, and by defining and classifying various circuit configurations as either implausible or incredible. Definitions of these two terms are essential to understanding the process:

Implausible: The term "implausible," when used in conjunction with a fire-induced circuit failure phenomenon, supports the PIRT panel's conclusion that the phenomena happening, while theoretically possible, would require the convergence of a combination of factors that are so unlikely that the phenomenon's occurrence can be considered statistically insignificant. In these cases, the PIRT panel could find no evidence of the phenomenon ever occurring, neither in operating experience nor during a fire test.

Incredible: The term "incredible," when used in conjunction with the phenomenon of a fire-induced circuit failure, signifies the PIRT panel's conclusion that the event will not occur. In these cases, the PIRT panel could find no evidence of the phenomenon ever occurring, and there were no credible engineering principle or technical argument to support its happening during a fire.

Circuit configuration not classified as either implausible or incredible are considered possible. The PIRT panel also developed consensus technical positions on several longstanding fire protection circuit issues. These positions were supported by data, engineering principles, physical configurations, manufacturers' input, operating experience, and expert judgment. Using the same definitions developed previously (implausible, incredible), the PIRT panel reached consensus technical positions[6] on the following items:

Implausible

- A single inter-cable hot short between
  - Two thermoset insulated cables
  - Thermoset insulated source[7] cable and a thermoplastic insulated target cable
- Two inter-cable hot shorts between two thermoplastic insulated cables

Incredible

- Inter-cable hot short between
  - Thermoplastic-insulated source cable and thermoset-insulated target cable regardless of the number of shorts needed to cause equipment spurious operation.[8]
  - Two inter-cable hot shorts between a thermoset-insulated source cable and a thermoplastic-insulated target cable
- Consequential three-phase AC short
- Consequential hot shorts in a DC compound-wound motor
- Multiple high impedance faults (MHIFs)
- Secondary fire concern for current transformers with turn ratios below 1200:5.

2.3 PRA expert elicitation process and preliminary results

The PRA expert elicitation panel was responsible for completing the final objective of this expert judgment effort, which is to develop best-estimates for the probability and duration of spurious operation given that cable damage happened due to fire. Specifically, the PRA panel was tasked with evaluating the probability distributions of the following main areas:

Probability[9] of spurious operation for:

- single-break solenoid operated valve (SOV) circuits,
- double-break SOV circuits,
- single break motor operated valve (MOV) circuits, and
- medium-voltage circuit breakers.

Probability of duration of spurious operation

- All circuit configurations

---

[6] *Several of these positions are based on underlying assumptions/conditions that need to be met for the particular issue to be dispositioned as either implausible or incredible (see Volume 1 of NUREG/CR-7150 [1] for details).*

[7] *The term "source cable" as used represents a cable that supplies energy to a "target cable" which receives energy and is associated with a device that can spuriously operate. These terms are only applicable to inter-cable failure modes.*

[8] *The PIRT panel believed that the difference in insulation robustness between a thermoset-insulated and a thermoplastic-insulated cable affects failure timing of cables. The panel determined that thermoplastic-insulated source to thermoset-insulated target configurations are incredible, while the failure characteristics of thermoset to thermoset-insulated cables should be considered implausible.*

[9] *All probabilities are conditional on cable damage due to fire*

To implement this effort, the project selected the expert elicitation process known as the Senior Seismic Hazard Analysis Committee (SSHAC) process. Using the SHAC process to represent the relevant informed technical community, the expert elicitation estimated the probability and duration distributions for all cable configurations, including those configurations where there was limited applicable test data. The SSHAC process is described in NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts" [12], and defines four levels of effort, with Level 1 being the most elementary (and least resource-intensive), and Level 4 being the most comprehensive (and most resource-intensive). The project management chose to conduct a Level 2 SSHAC process with inclusion of several Level 3 aspects, such as in-person panel workshops and the review by a participatory peer review panel (PPRP). Three in-person workshops were included to support communication among the panel members on complex issues, and the PPRP reviewed the technical aspects of the project as progress was made (as opposed to reviewing after the technical evaluations have been completed, when making changes is more difficult and resource-intensive), providing assurance that the proper process was followed. Specifically, the PPRP consisted of three experts from the NRC and the nuclear industry who were responsible for ensuring that the SSHAC process was followed. The primary focus of the PPRP was to ensure that the study incorporated the diversity of views prevailing within the technical community, that uncertainties were properly considered and incorporated into the analysis, and that the documentation of the study would be clear and complete.

Following the SSHAC process, the experts were classified into three categories, namely, evaluator experts, proponent experts, and resource experts. The evaluator experts constitute the technical integration (TI) team and are responsible for ultimately developing the composite representation of the informed technical community (called the community distribution) for each probability distribution for likelihood and duration of spurious operation. The role of a proponent expert is to advocate a specific model, method, or parameter for use in assessing the probabilities and durations of spurious actuations. The TI team evaluates the proposals developed by the proponent experts. Resource experts were responsible for presenting data in an impartial manner to inform the other panel members of the facts or technical understanding of the data. In an effort to provide continuity across the two panels, several of the experts on the electrical expert PIRT panel also served on the PRA panel.

Following two workshops and after having been provided input from the proponent and resource experts, the TI team developed the draft community distributions. A different approach was used for probabilities and for durations. For probabilities, this was accomplished in three steps. First, a beta distribution was fitted to the quantiles provided by each expert to obtain a probability distribution representing the expert's knowledge about a probability. In the second step, the TI team decided to use mathematical aggregation to combine the distribution from each expert into a single distribution. The linear opinion pooling (LOP) method was used, which is a weighted average of the individual distributions with weights wi summing to 1, as shown below.

$$f(\theta) = \sum_{i=1}^{n} w_i f_i(\theta) \tag{1}$$

where θ is an unknown quantity, such as the probability of spurious operation, fi(θ) is the individual distribution of expert i, n is the number of experts providing input, and f(θ) is the aggregated distribution.

In general, the TI team assigned equal weights to the individual proponent distributions. Since the distribution resulting from combing several distributions is typically not a parametric distribution, for convenience a beta distribution was fitted to the quantiles of the aggregated distribution, and the fitted distribution was considered the draft community distribution. Hence, the third step consisted of carrying out such fitting. Beta distributions were used in steps one and three because of two features of this type of distribution:

1) its range is the interval [0, 1], which corresponds to the values of probabilities; and

2) it is a flexible distribution that can take different shapes.

Figure 2 illustrates the last two steps, where PDF is defined as probability density function, and p(SO/fire) is the probability of spurious operation given fire damage. The dotted distributions are the distributions of four experts (Exp1 to Exp4), the thin continuous curve is the aggregated distribution resulting by applying the LOP method (previous equation), and the thick continuous distribution is the fitted distribution, that is, the draft community distribution (CD).

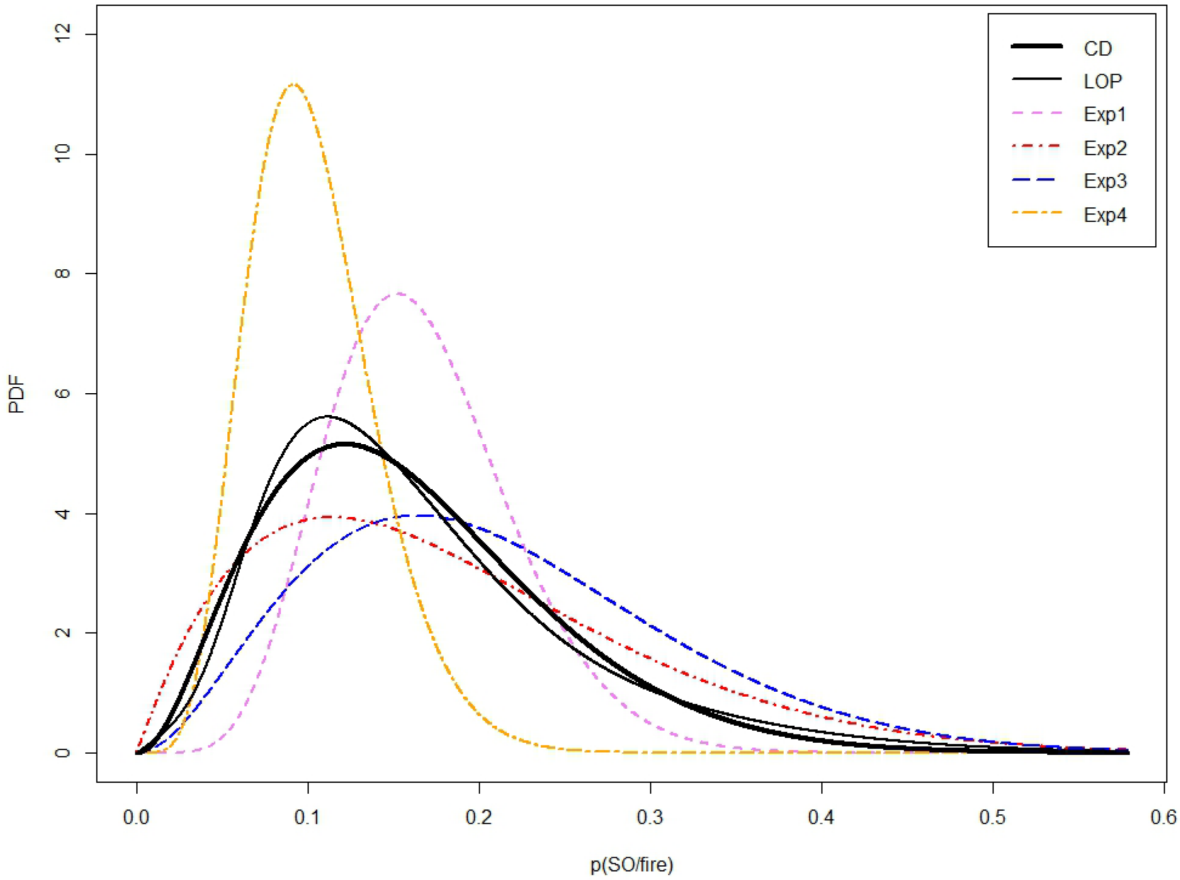


Figure 2. **An example of aggregating several individual distributions into a community distribution**

After considering the input from the proponent experts, the TI team selected the approach of carrying out a top-level evaluation for developing the draft community distributions for durations. In other words, a draft distribution for AC circuits and another for DC circuits were established, instead of a draft distribution for each combination of cable configuration (e.g., thermoset-insulated or thermoplastic-insulated cables) and failure mode of cables (e.g., intra-cable or inter-cable).

After the TI team develops the draft community distributions for probabilities and durations of spurious operations, a third and final SSHAC workshop was held. Workshop 3 focused on presenting and discussing the TI team's preliminary models and calculations in a forum that provides the opportunity for feedback by experts and the PPRP prior to finalization and documentation of the results. Following Workshop 3, the BNL moderator and the TI team finalized the calculations and documented the result in NUREG/CR-7150, Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure." The quantitative results are presented in Tables 1, 2, and Figure 3.

Table 1. **Summary of Mean Conditional Probabilities of Spurious Operation for Single Break Control Circuits**

Power Supply →			Grounded AC			Ungrounded AC (w/ Individual CPTs)			Ungrounded DC (or Ungrounded Distributed AC)			
			Conductor Hot Short Failure Mode									
Target Cable Configuration		Device Type	Intra-Cable	Inter-Cable	Aggregate	Intra-Cable	Inter-Cable	Aggregate	Intra-Cable	Inter-Cable	Ground Fault Equivalent	Aggregate
			1	2	3	4	5	6	7	8	9	10
Thermoset-Insulated Conductor Cable	1	SOV	0.42	0.01	0.43	0.64	9.7E-04	0.64	0.46	6.3E-03	0.17	0.56
		MOV	0.27	8.8E-03	0.28	0.38	8.5E-04	0.39	0.31	5.6E-03	0.11	0.40
		Circuit Breaker							0.40	6.3E-03	0.17	0.40
Thermoplastic-Insulated Conductor Cable	2	SOV	0.42	0.025	0.44	0.64	0.015	0.64	0.46	0.02	0.15	0.55
		MOV	0.27	0.022	0.29	0.38	0.013	0.39	0.31	0.018	0.10	0.40
		Circuit Breaker							0.40	0.02	0.15	0.40
Metal Foil Shield Wrap Cable	3	SOV	0.24	Incredible	0.24	0.54	Incredible	0.54	0.48	Incredible	0.30	0.63
		MOV	0.16		0.16	0.37		0.37	0.31		0.22	0.46
Armored Cable	4	SOV	0.047	Incredible	0.047	0.45	Incredible	0.45	0.73	Incredible	0.48	0.86
		MOV	0.034		0.034	0.27		0.27	0.45		0.29	0.61

Table 2. **Summary of Mean Conditional Probabilities of Spurious Operation for Ungrounded Double Break Control Circuits**

Target Cable Configuration		Power Supply Configuration	Device Type	Combinations of Conductor Hot Short Failure Modes: Intra-Cable & Intra-Cable (1)	Intra-Cable & Inter-Cable (2)	Inter-Cable & Inter-Cable (3)	Intra-Cable & Ground Fault Equivalent (4)	Inter-Cable & Ground Fault Equivalent (5)	Aggregate (6)
Thermoset-Insulated Conductor Cable	1	AC w/CPTs	SOV	0.43	0.065	Incredible			0.47
			MOV	0.30	0.065				0.35
		DC (AC w/o CPTs)	SOV	0.23	2.9E-03		0.077	Incredible	0.29
			MOV	0.16	2.9E-03		0.054		0.20
Thermoplastic-Insulated Conductor Cable	2	AC w/CPTs	SOV	0.43	0.082	5.3E-03			0.48
			MOV	0.30	0.082	5.3E-03			0.36
		DC (AC w/o CPTs)	SOV	0.23	9.5E-03	8.6E-04	0.07	3.1E-03	0.29
			MOV	0.16	9.5E-03	8.6E-04	0.049	3.1E-03	0.21
Metal Foil Shield Wrap Cable	3	AC w/CPTs	SOV	0.33	0.12	Incredible			0.41
			MOV	0.23	0.12				0.33
		DC (AC w/o CPTs)	SOV	0.27	Incredible		0.14	Incredible	0.36
			MOV	0.19			0.10		0.26
Armored Cable	4	AC w/CPTs	SOV	0.23	0.16	Incredible			0.35
			MOV	0.16	0.16				0.29
		DC (AC w/o CPTs)	SOV	0.55	Incredible		0.35	Incredible	0.70
			MOV	0.38			0.25		0.53

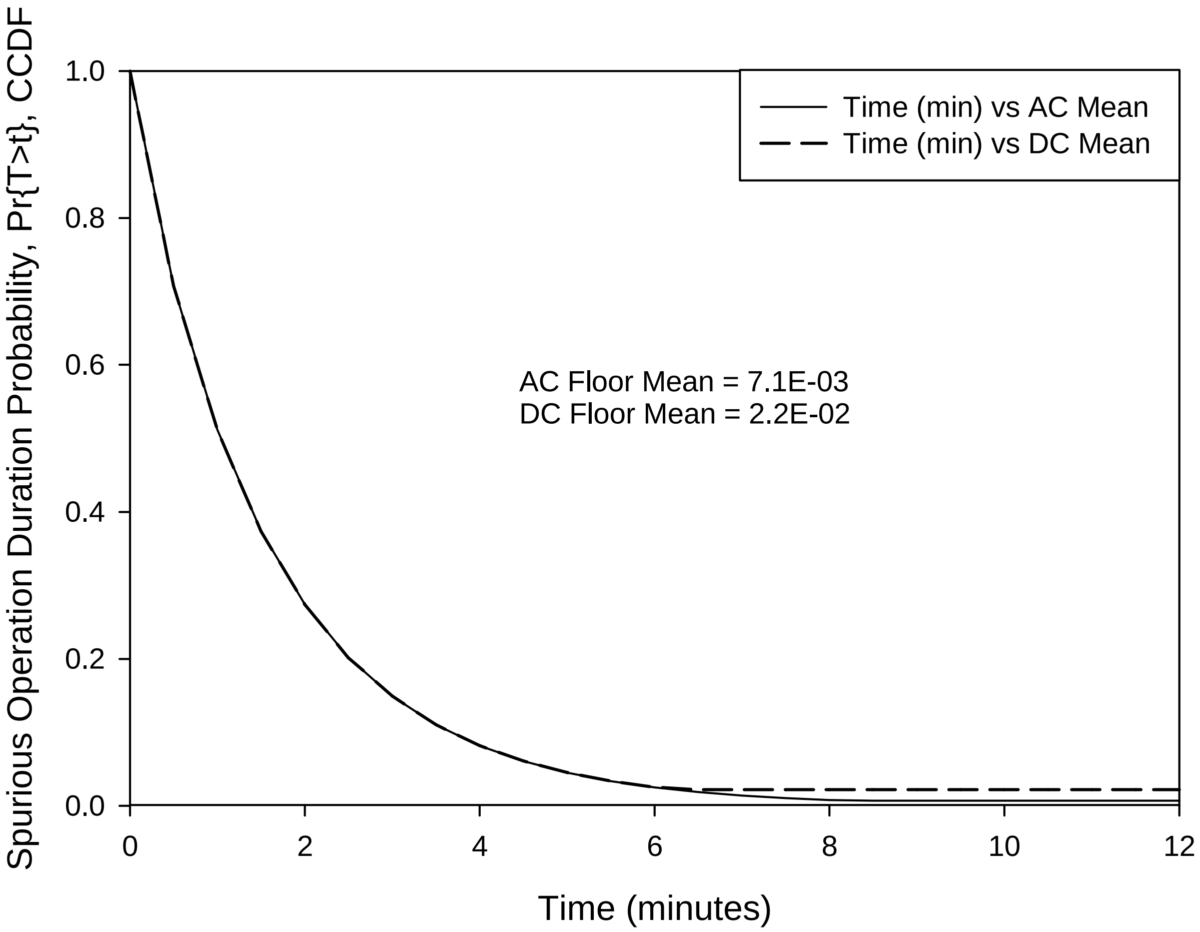


Figure 3. **AC and DC Spurious Operation Duration Mean Conditional Probability Plots**

3. Examples of Conditional Risk Changes

This section provides three simplified case studies to illustrate the changes in conditional risk estimation between the original NUREG/CR-6850 Task 10 method and the new quantification estimate developed using expert judgment in the JACQUE-FIRE program.

Case 1a – Solenoid Operated Valve (SOV)

- Electrical schematic shown in Figure 4
- Failure mode of concern is a hot short induced spurious operation causing valve to spurious open
- Cable B is a multi-conductor thermoset-insulated cable containing electrical nodes (P00, N00, R00, G00, SV0, SV1, and a spare), located in a steel ladder back cable tray.

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-2	0.62
NUREG/CR-7150	Table 1 (in this paper)	0.56
Conditional Risk Change		0.04 decrease (~6% risk reduction)

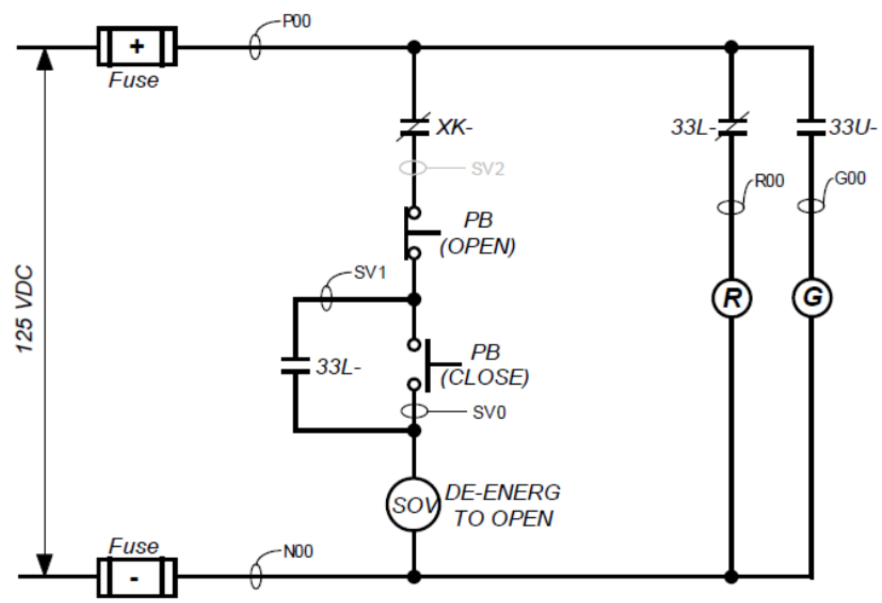


Figure 4. **Electrical Schematic for Typical SOV Circuit**

Case 1b – Solenoid Operated Valve (SOV)

- Same as Case 1a, except cable is armored.

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-5	0.15
NUREG/CR-7150	Table 1 (in this paper)	0.86
Conditional Risk Change		0.71 increase (~factor of 6 risk increase)

Case 2a – Motor Operated Valve (MOV)

- Electrical Schematic Shown in Figure 5
- Failure mode of concern is a hot short induced spurious closure of valve
- Cable B is a multi-conductor thermoplastic containing electrical nodes (X00, S01, R00, G00, SC1, and two spares), located in a steel ladder back cable tray.

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-3	0.32
NUREG/CR-7150	Table 1 (in this paper)	0.29
Conditional Risk Change		0.03 decrease (~10% risk reduction)

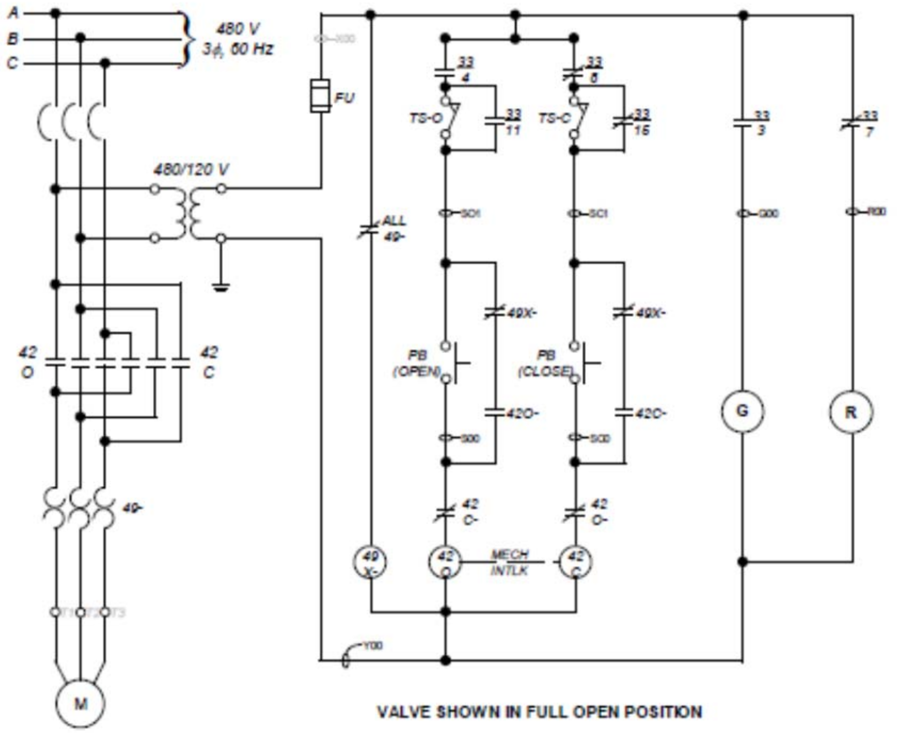


Figure 5. **Electrical Schematic for Typical MOV Circuit**

Case 2b, Motor Operated Valve (MOV)

- Same as Case 2a, except cable is armored.

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-5	0.075
NUREG/CR-7150	Table 1 (in this paper)	0.034
Conditional Risk Change		0.041 decrease (~50% risk reduction)

Case 3a – Double Break DC MOV

- Electrical schematic is shown in Figure 6
- Failure mode of concern is spurious opening (Raise) of valve while in remote
- Cable B is a multi-conductor cable thermoplastic insulated cable containing electrical notes (P01, N01, F02, F03, and a spare) located in a steel ladder back cable tray.

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-4	0.62
NUREG/CR-7150	Table 1 (in this paper)	0.21
Conditional Risk Change		0.41 decrease (~factor of 3 risk reduction)

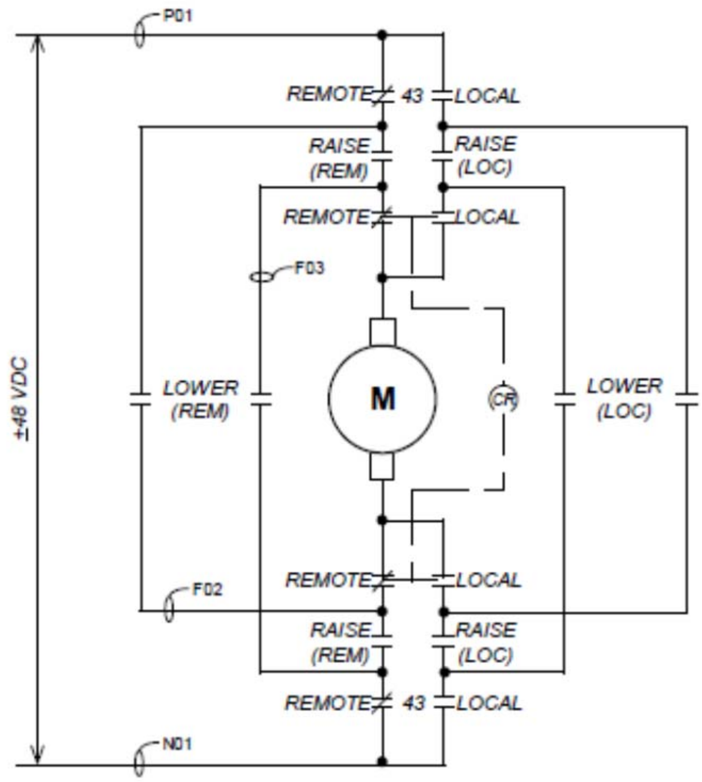


Figure 6. **Electrical Schematic for Double-Pole-Isolation DC MOV Control Circuit**

Case 3b Double Break DC MOV

- Same as Case 3a, except cable B is located in conduit

Results

Method	Data Source	Conditional Likelihood
NUREG/CR-6850	Table 10-5	0.16
NUREG/CR-7150	Table 1 (in this paper)	0.21
Conditional Risk Change		0.05 decrease (~30% risk reduction)

4. Conclusions

Data obtained from operating experience and tests are the main basis for developing estimates of probabilities and durations of fire induced circuit failures to be used in a PRA. For circumstances where limited, or no data, available, or where a consensus position on a technical issue is desired, the use of expert judgment can be a valuable tool. The use of two expert panels as described have shown, the PIRT and SSHAC methods in combination are useful for obtaining expert judgment on complex issues, in this case, to solve some of the problems faced in fire PRA.

The electrical expert PIRT panel results have provided technical consensus on several long time contentious safety issues related to fire PRA electrical circuit analyses. Specifically, the PIRT results categorized circuit configurations as being possible, implausible or incredible, which provided strong technical recommendations to support regulatory activities. Strong technical recommendations, as resulted from the two expert panels, provide knowledge for a better understanding of fire-induced hot short phenomena. In turn applying knowledge gained from such panels, enhance both deterministic and risk-informed fire analyses resulting in resolving long standing issues and implementing guidance to improve regulatory activities.

Additionally, the results of the PIRT panel identified areas where knowledge is low and suggested areas of future research prioritized based on risk insights. Finally, the PIRT results provided a strong technical basis and framework for the follow-on PRA expert elicitation panel to perform its analysis.

The PRA expert elicitation was conducted according to the SSHAC process, which provided the following main benefits: 1) well-defined roles for the experts engaged in the elicitation; 2) workshops with specific goals; and 3) a PPRP that reviewed and provided comments on the technical and process aspects of the elicitation. The continuous oversight of this PPRP team allowed for real time assurance that the SHAC process was being followed in order for the results to accurately represent the informed technical community The final output of this panel represents aggregated community distributions for the probabilities and durations of fire-induced spurious operations given cable damage due to fire.

The results of both panels are advancing the current state-of-the-art on probabilistic data for developing a Level-1 fire PRA. This data will provide more realistic results in fire PRAs based on test data supplemented by structured expert judgment. Further, these results will promote informed and stable regulatory decision making in U.S. NPP fire protection programs.

Acknowledgments

The authors of this paper would like to acknowledge those responsible for planning, conducting, supporting, and executing the work described in this paper. Those individuals are listed below.

Project Management
James Higgins (BNL)
Mark Henry Salley (NRC/RES)
Richard Wachowiak (EPRI)

Electrical Expert PIRT Project
Harold Barrett (NRC/NRR)
David Crane (Pyrolico Corporation)
Robert Daley (NRC/Region III)
Daniel Funk (Hughes Associates, Inc.)
Thomas Gorman (PPL Susquehanna, LLC)
Steven Nowlen (Sandia National Laboratories)
Andy Ratchford, (Ratchford Diversified)
Mano Subudhi (BNL)
Gabriel Taylor (NRC/RES)

PRA Expert Elicitation Project
Daniel Funk (Hughes Associates, Inc.)
Raymond H.V. Gallucci (NRC/NRR)
Dennis Henneke (GE Hitachi)
Gerardo Martinez-Guridi (BNL)
David Miskiewicz (EPM)
Steven Nowlen (Sandia National Laboratories)
Martin Stutzke (NRC/RES)
Mano Subudhi (BNL)
Gabriel Taylor (NRC/RES)
Kiang Zee (ERIN)

Participatory Peer Review Panel (PPRP)
Robert Cavedo (CENG)
Nathan Siu (NRC/RES)
Jing Xing (NRC/RES)

Project Support
David Gennardo (NRC/RES)
Nicholas Melly (NRC/RES)
Tammie Pennywell (NRC/RES)
David Stroup (NRC/RES)

References

[1] Subudhi, M., Higgins, J.,et.al., "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE),Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," Brookhaven National Laboratory, BNL-NUREG-98204-2012, EPRI 1026424, Nuclear Regulatory Commission NUREG/CR-7150 (October 2012).

[2] EPRI/NRC-RES, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Electric Power Research Institute TR 1011989, Nuclear Regulatory Commission NUREG/CR-6850 (2005, September).

[3] Subudhi, M., Literature Review of Environmental Qualification of Safety-Related Electric Cables," Brookhaven National Laboratories BNL-NUREG-52480 Vol. 1, Nuclear Regulatory Commission NUREG/CR-6384 Vol. 1 (1996, April).

[4] Kassawara, R., "Characterization of Fire-Induced Circuit Faults, Results of Cable Fire Testing," Electric Power Research Institute TR 1003326 (2002, December).

[5] Kassawara, R., "Spurious Actuation of Electrical Circuits Due to Cable Fires, Results of an Expert Elicitation," Electric Power Research Institute TR 1006961 (2002, May).

[6] Nuclear Regulatory Commission. Regulatory Issue Summary 2004-03, "Risk-Informed Approach for Post-Fire Safe-Shutdown Associated Circuits Inspections," (2004, March).

[7] Nowlen, S.P., Wyant, F.J., "Cable Response to Live Fire (CAROLFIRE)," Sandia National Laboratories SAND2007-600, Nuclear Regulatory Commission NUREG/CR-6931 (April 2008).

[8] Nowlen, S.P., Brown, J.W., Olivier, T.J., Wyant, F.J., "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," Sandia National Laboratories SAND2012-0323P, Nuclear Regulatory Commission NUREG/CR-7100 (April 2012).

[9] Diamond, D.J., "Expert Panel Report on Proactive Materials Degradation Assessment," Brookhaven National Laboratory BNL-NUREG-77111, Nuclear Regulatory Commission NUREG/CR-6923 (2007, February).

[10] Olivier, T.J. and Nowlen, S.P., "A Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire Modeling Applications," Sandia National Laboratories SAND2008-3997P, Nuclear Regulatory Commission NUREG/CR-6978 (2008 July).

[11] Taylor, G.J., Melly, N.B., Woods, H.R., Pennywell, T., Olivier, T.J., Lopez, C., "Electrical Cable Test Results and Analysis During Fire Exposure (ELECTRA-FIRE), A consolidation of Three Major Fire-Induced Circuit and Cable Failure Experiments Performed Between 2001 and 2011," Nuclear Regulatory Commission Draft NUREG-2128, (June 2012).

[12] Budnitz, R.J., Apostolakis, G., Boore, D.M., Cluff, L.S., Coppersmith, K.J., Cornell, C.A., Morris, P.A., "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," Lawrence Livermore National Laboratory UCRL-ID-122160, Nuclear Regulatory Commission NUREG/CR-6372 (1997 April).

[13] NRC. NUREG/CR-6834, "Circuit Analysis – Failure Mode and Likelihood Analysis," September 2003.

Predicting the Heat Release Rate of Liquid Pool Fires Using CFD

Topi Sikanen
VTT, Finland

Simo Hostikka
Aalto University, Finland

Abstract

Fires in nuclear power plants (NPPs) are a serious hazard for overall safety of the plant. One typical fire scenario in NPPs is a pool fire of e.g. transformer oil. It is therefore important to understand the dynamics of such fires and be able to predict their heat release rates with sufficient accuracy. Since fire safety analyses are increasingly done with the aid of computational fluid dynamics (CFD) models, there is a need to develop pool fire models to be used as boundary conditions in CFD simulations.

Predicting the heat release rates (HRRs) of pool fires is a difficult task. In CFD simulations, the pool itself is often modelled as a prescribed fuel inlet condition, with the burning rate obtained from experimental data. This approach is limited to situations where experimental data of burning rates is available. Effects of varying oxygen concentration and external heat sources are particularly tricky to include. When the liquid phase is included in the model, the in-depth heat transfer within the liquid phase is often neglected.

In this paper a model for predicting the heat release rates of liquid pool fires is presented. The model accounts for the in-depth heat transfer by both radiation and convection. The in-depth radiation transport is solved by a one dimensional radiation transport model together with effective absorption coefficients determined from experimental data. Possibility of convective heat transfer in the liquid phase is studied using effective thermal conductivity. The model is implemented as a boundary condition in the Fire Dynamics Simulator.

Results from simulations accounting for in-depth radiation transport and convective motions are compared with experimental data. The importance of various modelling assumptions is explored along with the sensitivity of the models to input parameters. It is shown that the in-depth heat transfer has an effect on the dynamics of the fire. However for maximum burning rates, the most important parameters are found to be related to gas phase combustion. Overall, the predictions of maximum burning rates are found to be in agreement with experimental data.

1. Introduction

Pool fires are an important class of industrial fire hazards because substantial amounts of flammable liquids are present in most industrial facilities, and because the rapid development of the heat release rate in such fires poses a challenge to the safety systems. Pool fires have been studied for decades and this work has been collected in several review articles [1,2,3]. The focus of the research has usually been on the steady state behavior and maximum burning rates of pool fires of various sizes. The result of such study is often a empirical correlation for the burning rate of a pool fire. A recent example is the study by Ditch et al. [4] where the authors correlated the mass burning rate with the fuel heat of gasification and smoke point.

The fire analyses are often carried out using the Computational Fluid Dynamics (CFD) type of fire simulations. The most important boundary condition for these simulations is usually the pool burning rate. While many of the analyses can be performed by prescribing the pool burning rate using either experimental data or empirical correlations as sources of information, there are situations where the conditions of the fire scenario are so much different from any experimental study that a reliable prediction of the pool burning rate cannot be made in advance. Examples of significant conditions are the ambient temperature and radiation level, side wind, oxygen vitiation. On the other hand, the heat transfer conditions within the pool itself can be significantly different from the empirical conditions. Furthermore, the transient nature of the analyses requires the knowledge on the time-dependent burning rate, not just the peak or steady state value. It is therefore necessary to develop sub-models for the CFD fire models that can predict the pool fire dynamics and burning rate during the simulation.

Predictive CFD simulation of the pool burning rate was previously performed by Hostikka et al. [5]. In their model, the liquid evaporation rate was calculated iteratively over the course of the simulation to maintain an equilibrium fuel vapor pressure in the first gas-phase cell above the liquid boundary. The heat transfer inside the liquid layer was calculated using a one-dimensional heat conduction solver. In the results, only the steady state burning rate value was observed paying no attention to the temporal development.

The vast majority of the pool fire studies have focused on the global burning rate, gas phase conditions and the flame heat fluxes. The question of heat transport within the fuel has received less attention. Vali et al. [9] noted that in their experiments there was a region of near constant temperature under the surface of a burning liquid. In this layer the heat transfer was driven by convection caused by the hot walls of the pool.

Table 1. **Thermophysical properties of liquids considered in this paper**

Fuel	**ρ (kg/m3)**	**λ (W/mK)**	**c_p (kJ/kg)**	**Δh_c (MJ/kg)**	**Δh_v (kJ/kg)**	**χ_r (-)**	**Y_s (-)**	**κ (1/m)**
Acetone	792.5	0.18	1.65	30.5	501	0.27	0.014	100
Benzene	876.5	0.167	1.74	39.9	393	0.32	0.181	123
Butane	584	0.124	2.28	45.7	362	0.3	0.029	100
Ethanol	789	0.17	2.44	26.8	837	0.18	0.008	1534.3
Heptane	684	0.14	2.25	44.4	365	0.35	0.037	187.5
Methanol	791.8	0.21	2.48	19.8	1099	0.18	0.001	1520

2. Mathematical Model

The simulation tool used in this work is the Fire Dynamics Simulator (FDS) [10]. FDS solves the Navier-Stokes equations in a form suitable for low-Mach number thermally driven flows. Turbulence is treated by Large Eddy Simulation (LES). In this section, a description of the FDS liquid pyrolysis model is presented.

FDS solves a one dimensional heat conduction equation for the liquid fuel

$$\rho c \frac{\partial}{\partial} = \frac{\partial}{\partial x} \lambda \frac{\partial T}{\partial x} + \dot{q}'''$$

$$-\lambda \frac{\partial T}{\partial x}\Big|_{x=0} = h\left(T_g - T_s\right) - \Delta h_v \dot{m}'' \tag{1}$$

$$h = \max\left[1.52\left|T_g - T_s\right|^{\frac{1}{3}}, \frac{\lambda}{1} 0.037 Re^{4/5} Pr^{1/3}\right]$$

Here *ρ*, *c, λ and T* are respectively the fuel density, specific heat, thermal conductivity and temperature. The subscripts S and g refer to conditions at the fuel surface and in the first gas phase cell respectively. On the surface Δh_v and $\dot{m}''$ are the heat of vaporization and the evaporation mass flux, respectively.

The radiative transport can be described as volumetric heat-source term $\dot{q}'''$ in Equation 1. The FDS condensed phase model uses a "two-flux" model, where the radiative intensity is assumed to be constant in "forward" and "backward" hemispheres. The forward radiative heat flux into the fuel is

$$\frac{d\dot{q}^+}{dx} = \kappa(\sigma T^4 - \dot{q}^+) \tag{2}$$

A corresponding formula can be written for the backward flux$\dot{q}^-$. The heat source term in Equation 1 is the difference between the forward and backward fluxes

$$\dot{q}''' = \frac{d\dot{q}^+}{dx} - \frac{d\dot{q}^-}{dx} \tag{3}$$

Boundary condition at the fuel surface is given by

$$,\dot{q}^+|_{x=0} = \dot{q}''_{in} + (1-\varepsilon)\dot{q}^- \tag{4}$$

where **ε** is the fuel emissivity and $\dot{q}''_{in}$ is the incoming radiative flux.

The effect of the unresolved concentration boundary layer near the pool surface is taken in to account. In this model the mass flux is given by [11]

$$\dot{m}'' = h_m \rho_{f,g} log\left(\frac{X_G - 1}{X_f - 1}\right); X_s = exp\left[-\frac{\Delta h_v W}{R}\left(\frac{1}{T_S} - \frac{1}{T_b}\right)\right] \tag{5}$$

Here $h_m = Sh\mu_g/Sc\Delta x$ is the mass transfer coefficient and $\rho_{f,g}$ and X_G are the density of the fuel vapour and the volume fraction of fuel vapour in the grid cell adjacent to the pool surface, W is the molar mass of the fuel gas and R is the universal gas constant. The Schmidt number Sc is 1 and the Sherwood number is given by

$$Sh = 0.037 Sc^{\frac{1}{3}} Re^{\frac{4}{5}};\ Re = max\left[5 \times 10^5, \frac{\rho u L}{\mu}\right]\ . \tag{6}$$

The Reynolds number is calculated based on conditions in the cell adjacent to the surface. Properties of liquids considered in this paper are listed in Table 1. In the above, L is some characteristic length that is set to 1 meter for calculations in this paper. Note that the Reynolds number is bounded from below. This ensures a non-zero mass flux from liquid fuels and thus circumvents the need to model the ignition process. The Reynolds number varies over time through the gas speed dependence, and so do the Sherwood and mass transfer numbers. We will show later that the predicted burning rates are relatively insensitive to the mass transfer coefficient.

3. Effective Absorption Coefficients for In-Depth Radiation Absorption

Absorption of thermal radiation in semitransparent media is highly dependent on the wavelength of the radiation and cannot be represented by a single number for all cases. The absorption coefficients in Table 1 are based on a curve fitting procedure. The procedure is as follows:

- Start with spectrally resolved absorption coefficients κ_v for a liquid. Here v is frequency.
- Assume incoming radiation follows blackbody spectrum with temperature 1450 K

$$I_{v,in} = \frac{2hv^3}{c^2} \frac{1}{exp\left[\frac{hv}{kT}\right] - 1} \tag{7}$$

- Calculate transmitted fraction of radiation at distance S from the liquid surface by line by line application of beers law.

$$I(S) = \int_0^\infty I_{v,in}\, exp[-\kappa_v S] dv \tag{8}$$

- Optimize to find a value for absorption coefficient κ that minimizes some error metric between the predicted flux $\dot{q}^+$ from solution of Eq. 2 and the flux calculated from Eq. 8.

Two choices need to be made in this process:

- Choice of optimization metric: Do we want to reproduce the heat source distribution $\dot{q}'''$ accurately (Eq. 10 or do we want to match the flux $\dot{q}^+$ at certain distance from the surface (Eq. 9). The latter choice will lead to correct amount of energy being deposited in the liquid layer
- The path length L at which the flux $\dot{q}^+$ is matched or over which the heat source distribution $\dot{q}'''$ is matched.

These choices lead to either

$$\kappa = \arg\min\left[\frac{\dot{q}^+(L)}{\dot{q}^+(0)} - \frac{I(L)}{I(0)}\right] \tag{9}$$

or

$$\kappa = \arg\min\left[\int_0^L \left(\frac{\dot{q}^+(x)}{\dot{q}^+(0)} - \frac{I(x)}{I(0)}\right)^2 dx\right]. \tag{10}$$

With the exception of acetone, butane and methanol, the absorption coefficients in Table 1 are based on application of Eq. 10 with a path length L of 3 mm. This produces a fairly realistic heat source distribution near the surface. However, the fraction reaching the backside of the pool is likely to be too small. In this paper we will explore the effect of this choice by varying the calculated absorption coefficients.

Absorption data for butane and acetone was not available and the values are guessed. For methanol absorption coefficient spectra was available, but the assumption of blackbody radiation is not applicable for low-sooting fuels such as methanol. Instead, the value for methanol is based on the absorption coefficient of ethanol. The absorption coefficient for ethanol in turn is calculated with the help of measured emission spectrum of ethanol flame data of Suo-Anttila et al. [12].

4. Modelling the Effects of Convection in the Liquid Phase

There are several sources of motion within fuel. One source is uneven burning rate of fuel. This will cause a Fuel to flow towards regions of high burning rates. Second source are the hot walls of the pool where heat transferred through the pool walls creates natural convection currents. Third source is the in-depth radiation absorption. During pool combustion, the pool surface is cooled by evaporation. The heat to the surface is provided by conduction and convection from both the liquid phase and gas phase. Therefore the liquid side convection can have an important effect on the heat balance on the liquid surface.

Accurately modeling all these phenomena would require solving the full Navier Stokes equations for the liquid phase together with heat transfer modeling for the pool walls. Such approach is too time-consuming for practical work and too complex for use in engineering calculations. Instead, a simplified model is sought. We try to model the overall effective heat transfer through the whole fuel layer.

Since Eq. 1 does not explicitly contain convective heat transfer within the pool the easiest way to include the effect of convection is to increase the thermal conductivity λ. Values for effective thermal conductivity could be sought in terms of Nusselt number correlations

$$Nu = \frac{\lambda_{eff}}{\lambda} \tag{11}$$

Such correlations are usually given in terms dimensionless numbers such as the Rayleigh number. In general convective motions can increase the heat flux through a fluid layer by several magnitudes. In this work we simply assume that the Nusselt number is O (10).

5. Model Validation

5.1. Maximum burning rates

To expand our selection of fuels and pool sizes. A series of large pool fires is simulated. These simulations consider large 1m x 1m rectangular pools. A 5 cm discretization interval is used for the mesh and the size of the computational domain is 1.6 m × 1.6 m × 4.8 m. Unless otherwise stated, there is a 5 cm lip on the pools. Computational domain is illustrated in Figure 1.

Once more we make use of empirical correlations to validate the results. The maximum burning rates of liquid pool fires are well correlated with

$$\dot{m} = 1 \times 10^{-3} \frac{\Delta h_c}{\Delta h_g}; \qquad \Delta h_g = \Delta h_v + \int_{T_0}^{T_b} c_p dT. \tag{12}$$

The predicted burning rates are compared with experimental correlations. A well-known correlation by Babrauskas [1] gives the mass burning rate per unit area as a function of pool diameter D

$$\dot{m}'' = \dot{m}''_{\infty}(1 - exp[-k\beta D]). \tag{13}$$

Recently Ditch et. al. [4] derived a empirical correlation for the heat flux incident on pool surface given by

$$\dot{q}'' = \dot{m}''\Delta h_g = 12.5 + 68.3Y_s^{\frac{1}{4}}\left\{1 - exp\left(-\left[\tfrac{4}{3}\Delta h_g D\right]^{\frac{3}{2}}\right)\right\}. \qquad (14)$$

Largest differences in mass loss rate between simulation predictions and Eq. 15 are seen for heptane and butane. For butane, using correct value for the absorption coefficient could result in better predictions. For benzene Eq. 16 predicts a very high mass loss rate. This is due to the very high soot yield of benzene. Ditch et al. [4] noted that there is very little data for large benzene fires but for a 6m pool fire the value 0.09 kg/m2s has been reported. This value agrees well with the predictions of FDS and Eq. 15.

Fig. 1 shows a comparison of FDS predicted burning rates and those predicted by the Babrauskas correlation and the Ditch et. al.. correlation (Eqs. 13 and 14). In most cases the burning rates predicted by FDS are lower than those predicted by Equation 2.

There is considerable scatter in the literature for mass burning rates of heptane. This is evidenced by the large confidence bands for the Babrauskas correlation. The Ditch et. al. (Eq. 14) correlation gives a value close to the upper limit of the confidence band while FDS corresponds to the lower limit. Several reasons for the scatter in burning rates have been proposed. These reasons include varying lip heights, fuel level control and pan shape.

The predictions for butane have largest scatter of all fuels considered here. Due to its low boiling point, butane is likely to be boiling in any experiments. The evaporation model used in this paper does not include boiling and thus is likely to lack some essential physics for predicting butane pool fires. The differences in predictions from Eq. 13 and 14 are also largest for butane. There is very little data on butane pool fires in the open literature and as such the difference could be simply due to different datasets.

Notably Eq. 14 predicts burning rates that are very close to the line defined by Eq. 12. With the exception of butane, the burning rates predicted by FDS follow the same trend. This highlights the importance of the thermal properties of fuels in predicting burning rates.

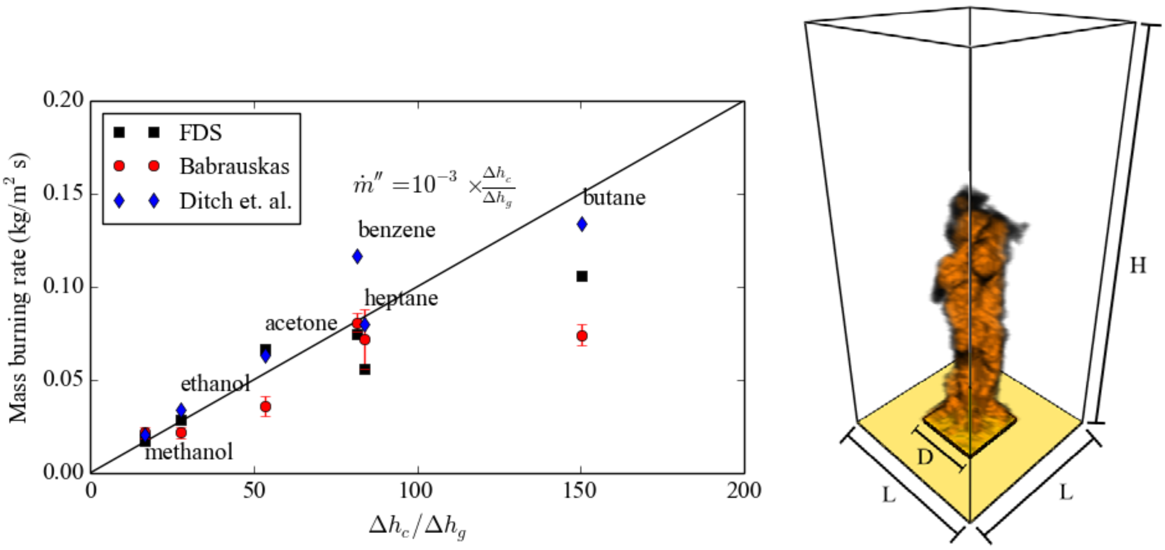


Figure 1. **Illustration of the computational model (right) and results from validation (left)**

5.1.1. Sensitivity analysis

For buoyancy dominated flows, such as pool fires, the adequacy of the grid resolution can be assessed using the Resolution Index. The RI is defined as

$$RI = \frac{D^*}{\Delta x}\,;\; D^* = \left(\frac{\dot{Q}}{\rho_\infty c_p T_\infty \sqrt{g}}\right)^{\frac{2}{5}},\ \text{and}\ \dot{Q} = \dot{m}'' A \Delta h_c. \tag{15}$$

Here Δx is the grid resolution, ρ_∞ T_∞, c_p and g are the ambient density and temperature, specific heat of air and the gravitational acceleration respectively. In the equation for heat release rate $\dot{Q}$, Δh_c is the heat of combustion of the fuel gas and A is the surface area of the fuel pan. Values of resolution index over 10 have been found to produce adequate results for predictions of flame heights.

Fig. 2 shows the burning rates of heptane pool fires with varying widths and three different grid resolutions. Note that in these simulations there was no lip on the pool and thus the burning rates are somewhat different from those presented in Fig. 1. The lip was left out of the model because the changing grid resolution would have necessarily led to a changing lip height in the computations, thus adding to the grid sensitivity. Coarse grids (lower values of RI) lead to higher predicted burning rates. For the finest grid the maximum burning rate agrees well with the correlation of Babrauskas (Eq. 13).

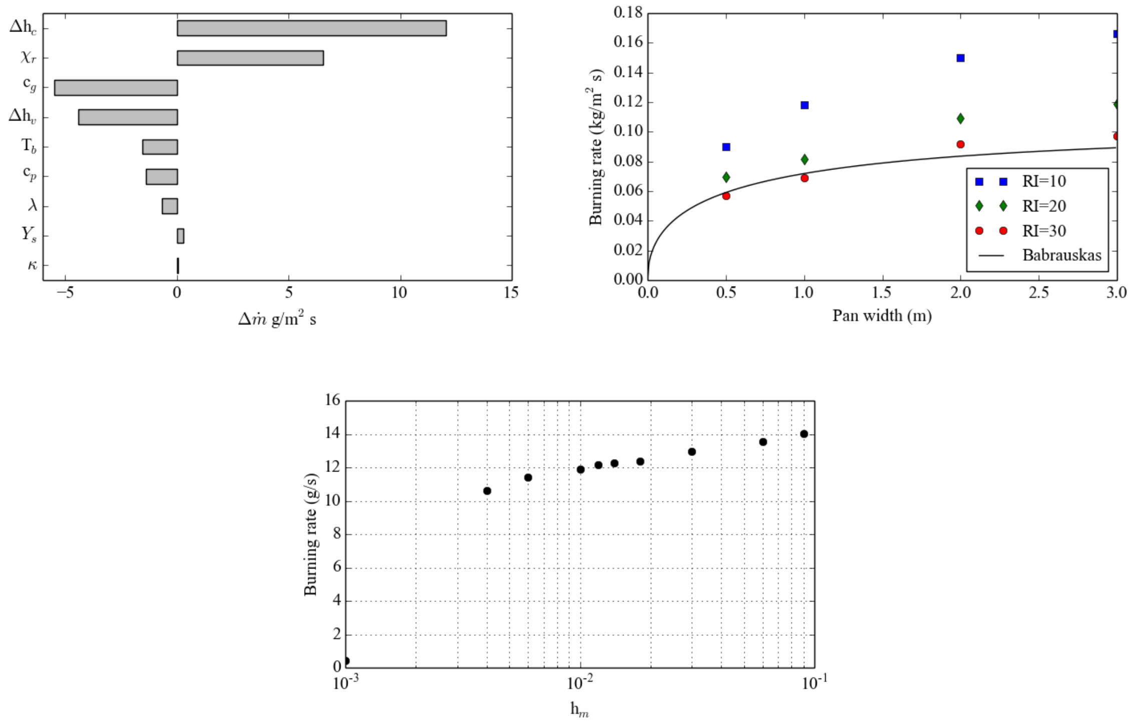


Figure 2. **Sensitivity of maximum burning rate of a 1 m² heptane pool fire to model parameters (left). Sensitivity to grid resolution for various sizes of heptane pools (right). On the bottom, sensitivity of burning rate of an ethanol pool fire to the value of mass transfer coefficient h_m.**

Fig. 2 also presents the model sensitivity of a $1 m^2$ heptane pool fire to different thermophysical parameters. Variables considered are the heat of combustion Δh_c radiative fraction χ_r specific heat of the fuel vapor c_g heat of vaporization Δh_v, the boiling temperature T_b, specific heat of fuel liquid c_p, thermal conductivity of the fuel liquid λ, soot yield Y_s, and the absorption coefficient κ. The sensitivity analysis was carried out by varying each variable by 10% and consequently estimating the sensitivities by central finite differences.

Values in Fig. 2 are arranged in decreasing order by their magnitude. The most important parameters turn out to be related to gas phase properties. The heat of combustion and the radiative fraction control the source term in the radiative transfer equation and consequently the amount of radiative feedback to the fuel surface. Increasing either of these variables will increase the burning rate. The third most important variable is the specific heat of gas. Increasing the specific heat of the vapors leads to lower burning rates. The bottom of Fig. 2 shows the effect of mass transfer coefficient on the predicted burning rate for a ethanol pool fire. It can be seen that after some threshold value, the mass transfer coefficient has a very modest effect on the result: An order of magnitude change leads to a change of only few percent in the predicted burning rate.

The findings here are in line with Eqs. 15 and 16 that only include the thermal properties of the fuels. The maximum burning rates of large pools are mostly related to thermophysical properties of the fuel. Most important parameters are heat of combustion, specific heat of the fuel gas and the heat of gasification of the fuel.

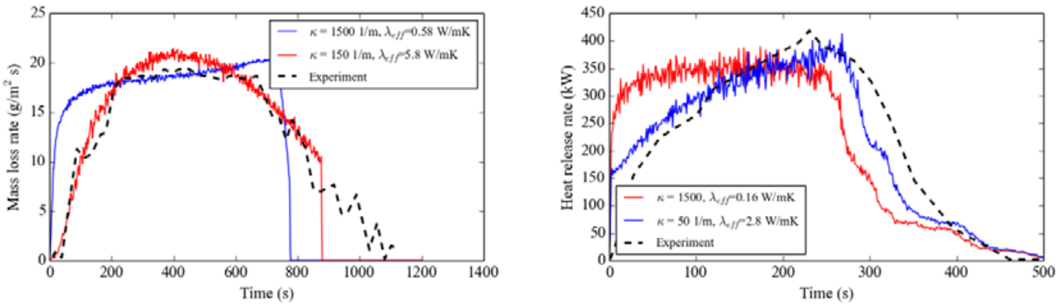


Figure 3. **Evaporation of water under constant heat flux (left) and burning rate of a ethanol pool fire.**

5.1.2. Pool fire dynamics

Up to this point we have focused on the prediction of maximum burning rates, ignoring the detailed dynamics of the fires. We discovered that the maximum burning rate of a liquid pool fire is mostly dependent on the thermal properties of the liquid fuel and its vapor. In this section we will see that the heat transfer within the fuel plays a role in determining the dynamics of a pool fire.

Fig. 3 shows a plot of the vaporization rate of water in a $0.0072 m^2$ Pyrex glass dish against time at $50 kW/m^2$ of external heat flux, measured in the ASTM E2058 fire propagation apparatus [13]. The depth of the water is 1.4 cm. This validation case has the benefit of removing the combustion and thus removes the need to predict the heat flux incident on the surface. Fig. 3 also shows the HRR from a 4kg ethanol pool fire with 0.7 m and 0.8 m side length and depth of 1 cm [11].

Two cases are considered for both ethanol and water, in the first case, we employ the default values for the liquids from Table 1. In the second case we assume that the thermal conductivity is approximately 10 times its default value in accordance with the discussion from section 3. In addition, the absorption coefficient is decreased to allow radiation to reach the bottom of the liquid. Even though the layer thickness is similar for the water and the ethanol simulations, the spectrum of incoming radiation is different, leading to different effective absorption coefficients.

It can be clearly seen that changes in absorption coefficient and thermal conductivity mostly affect the dynamics of the mass loss rate curve and have only a modest effect on average and maximum evaporation rates. With default values, the burning and evaporation rates are almost constant. When the absorption coefficient is decreased and the thermal conductivity is increased, the dynamics of the evaporation curve are better captured. This suggests that the internal heat transfer in the liquid is important in describing the detailed dynamics of the evaporation curve.

6. Conclusions

This paper considered modeling of liquid pool fires and the effect of in-depth heat transfer in the liquid phase on predicted burning rates. Effective radiative absorption coefficients based on spectrally resolved data were used when available. Effect of in-depth convective heat transfer was modeled by an effective thermal conductivity in the one-dimensional conduction equation.

Grid refinement studies showed that the maximum burning rates can be predicted accurately when the gas phase equations are solved on a fine enough grid. The grid resolution has a large effect on the incident heat flux and consequently on the burning rate of the pool fires. Modeling of the mass transfer in the liquid-gas interface was found to have little importance in predicting the maximum burning rates or the dynamics of pool fires. The most important parameters in predicting the burning rate are related to the gas phase combustion. These results are in agreement with the vast literature on pool fires where the maximum burning rates of pool fires have been found to mostly depend on fire geometry and thermophysical parameters of the fuel gas.

In-depth heat transfer in the form of in-depth radiation absorption and enhanced heat transfer in the liquid due to convective motions may be important in predicting the detailed dynamics of the fire. The results indicate that the heat transfer from the bottom of the pool through the liquid to the pool surface plays a large role in the dynamics of a pool fire. This result agrees with the experimental results of Vali et al. [9].

All pool fires considered in this paper were in open atmosphere or in large rooms. Better models for the internal heat transfer and validation of burning rate predictions in confined situations are future research topics. However, the results indicate that the prediction of pool fire burning rates is mostly an issue of predicting the heat flux to the pool surface. Effects of the compartment walls and oxygen levels are then included through the heat flux predictions. Therefore, using this model in compartment fire calculations seems justified.

Acknowledgements

This work was partially funded by The State Nuclear Waste Management Fund (VYR).

References

[1] Babrauskas V. Estimating large pool fire burning rates. Fire Technology, 19(4):251–261, 1983.

[2] Joulain P. The behavior of pool fires: state of the art and new insights. In Symposium (International) on Combustion, volume 27, pages 2691–2706. Elsevier, 1998.

[3] Steinhaus T., Welch S., Carvel R., and Torero J. Large-scale pool fires. Thermal Science, 11(2):101–118, 2007.

[4] Ditch B., de Ris J., Blanchat T., Chaos M., Bill R., and Dorofeev S. Pool fires an empirical correlation. Combustion and Flame, 160(12):2964 – 2974, 2013.

[5] Hostikka A., McGrattan K., and Hamins A.. Numerical Modeling of Pool Fires using Large Eddy Simulation and Finite Volume Method for Radiation. In Fire Safety Science – Proceedings of the Seventh International Symposium, pages 383–394. International Association for Fire Safety Science, 2002.

[6] Suard S., Forestier M., and Vaux S. Toward predictive simulations of poolfires in mechanically ventilated compartments. Fire Safety Journal, 61:64–64, 2013.

[7] Hayasaka H. Unsteady burning rates of small pool fires. In 5th Symposiumon Fire Safety Science, pages 499–510, 1997.

[8] Chen B., Lu S, Li C., Kang Q, and Lecoustre V. Initial fuel temperature effects on burning rate of pool fire. Journal of Hazardous Materials, 188(13):369 – 374, 2011.

[9] Vali A.,Nobes D. and Kostiuk L. Transport phenomena within the liquid phase of a laboratory-scale circular methanol pool fire. Combustion and Flame, (0):–, 2013.

[10] McGrattan, K., Hostikka, S., McDermott, R., Floyd, J., Weinschenk, C., Overholt, K. 2013. Fire Dynamics Simulator Technical Reference Guide Volume 1: Mathematical model. National Institute of Standard and Technology. NIST Special Publication 1018, Sixth Edition

[11] Thomas, I. R., Moinuddin, K. A., & Bennetts, I. D. (2007). The effect of fuel quantity and location on small enclosure fires. Journal of Fire Protection Engineering, *17*(2), 85-102.

[12] Suo-Anttila, J. M., Blanchat, T. K., Ricks, A. J., & Brown, A. L. (2009). Characterization of thermal radiation spectra in 2m pool fires. Proceedings of the Combustion Institute, *32*(2), 2567-2574.

[13] Tewarson, A. 2008. Generation of Heat and Gaseous, Liquid, and Solid Products in Fires. In *SFPE Handbook of Fire Protection Engineering*, 4^{th} ed. Society of Fire Protection Engineers, the National Fire Protection Association, Quincy, MA. 3-1 – 3-59.

Predicting Fire Spread of a Cable Tray Using Methods of Pyrolysis Modelling

Anna Matala
VTT, Finland

Simo Hostikka
Aalto University, Finland

Abstract

Significant proportion of the fire load in NPP comes from the electrical cables. In this paper, the process of cable modeling is presented starting from small scale and ending up to the large scale validation. The modeling process includes experimental testing, material modeling and parameter estimation, and model validation. The numerical simulations are made using Fire Dynamic Simulator (FDS) and the modeling process is demonstrated using a PVC-cable and experimental results of U.S. NRC CHRISTIFIRE project. The two alternative reaction paths are made for the cable, and the models are validated in the large scale simulations. The benefits and targets of development of the modeling process and the methods involved will also be discussed. The results are promising, and the methods have already been used for predicting flame spread in a cable tunnel, as a part of PRA of a Finnish NPP.

1. Introduction

Significant proportion of the fire load in NPP comes from the electrical cables. The cable fires are not only important because of the heat and smoke they produce, but also because a failure of important cables may cause a malfunction of an important safety system. In order to predict the fire spread correctly in different conditions, the cable must be modeled starting from the thermal degradation (or pyrolysis) of the solid phase. The pyrolysis modelling process is described and demonstrated step-by-step by using experimental results of Christifire campaign.

In order to predict the fire spread correctly in different conditions, the cable must be modeled starting from the thermal degradation (or pyrolysis) of the solid phase. The rate of the degradation depends on the material temperature, defined by a set of three parameters. These parameters are not generally known, and they depend on the choices on the reaction path and model. Therefore they cannot be directly measured, and must be estimated from the small scale experimental results. The heat transfer and combustion are controlled by other material and model specific parameters that can either be measured or estimated from the bench scale experimental results. The material modeling requires many modeler choices on the reaction path, number of reactions and the overall level of complexity of the model. In addition, the decisions on the model geometry and layer structure are extremely difficult and important for a model of cylindrical, non-uniform materials like cables. There are two alternative means of modeling such an object; it can either be handled as a solid, rectangular block or a sub-grid scale particle. Both methods have their benefits and shortcomings [1].

In this paper, the process of cable modeling is presented starting from small scale and ending up to the large scale validation. The modeling process includes experimental testing, material modeling and parameter estimation, and model validation. The numerical simulations are made using Fire Dynamic Simulator (FDS) and the modeling process is demonstrated using a PVC-cable and experimental results of U.S. NRC CHRISTIFIRE project [2]. The two alternative reaction paths are made for the cable, and the models are validated in the large scale simulations. The benefits and targets of development of the modeling process and the methods involved will also be discussed.

The results of cable modeling using the presented methods are promising. The methods have already been used for predicting flame spread in a cable tunnel, as a part of PRA of a Finnish NPP [3]. In addition, the material modeling of cables and other materials have been used in several other studies and safety analyses as well.

2. Pyrolysis Modelling

2.1 Modelling process

All modelling in this paper has been done using Fire Dynamics Simulator version 6 [4]. The process scheme is shown in Figure 1. It consists of four main phases: Sample preparation, experimental work, modelling and parameter estimation, and model validation.

The Step 1, sample preparation, is to familiarize oneself with the sample material, and prepare it for testing. Cables consist of several components, and it is necessary to evaluate whether the components can be separated for small-scale testing or not. All the direct measurements should also be performed for separated samples whenever possible, in order to determine properties for pure components. If any information of the component materials can be found from the literature, it can be taken into account when the testing is planned.

The Step 2 is the experimental work. For modelling, the most important experiments are Thermogravimetric analysis (TGA) and cone calorimeter. However, additional experiments provide more information and are useful for building an accurate model. Good experiments are, e.g., Differential scanning calorimetry (DSC), Microscale combustion calorimetry (MCC) or any measurement of thermal properties. The small-scale experiment should be done to each separable component individually. TGA should be performed both in nitrogen and air in order to see the difference of the oxidative environment, and at several heating rates (2-30 K/min). The cone calorimeter experiments should be performed at several heat fluxes (25-75 kW/m^2) as well.

Step 3, modelling and parameter estimation, is the most challenging and important part of the process. It starts from the determination of the reaction path, including decisions about reaction types, number of reactions and residue materials. Literature information helps at this point, but it also seems that the larger scale results are not very sensitive to the choice of reaction path [1],[5]. In any case, the kinetic parameter values depend strongly on the chosen reaction path, and cannot be generally used with any other reaction path, and therefore the decision needs to be made in the beginning. The kinetic parameters (A, E, N) are typically estimated from TGA results (in some cases, also MCC has been used [6]). There are two basic methods for the parameter estimation: analytical methods and curve-fitting algorithms [1]. Analytical methods are based on reference points in the TGA data, and can be solved easily without any specific software. They are fast and robust for simple data, but may not work very well with many overlapping reactions or noisy data. Curve-fitting algorithms, in fire research especially evolutionary algorithms (e.g., Genetic algorithms (GA)), are not based on solving any equation but rather fit a curve to the experimental results by minimizing error between model and experiments. They are very effective with more complicated data and are not limited to any model, but the estimation takes longer time and requires knowledge and specific software from the modeller. In this work, all the parameter estimations have been made using GA. When the kinetic parameters have been successfully estimated, they are fixed, along with all other measured parameters (e.g., density, specific heat capacity). Then the cone calorimeter model is made, including decisions of the layer structure and other parameters. When the cone calorimeter model has been decided, all the missing parameters are estimated by fitting the model to cone calorimeter results. If the resulting parameters repeat the cone calorimeter results accurately, the material model is completed.

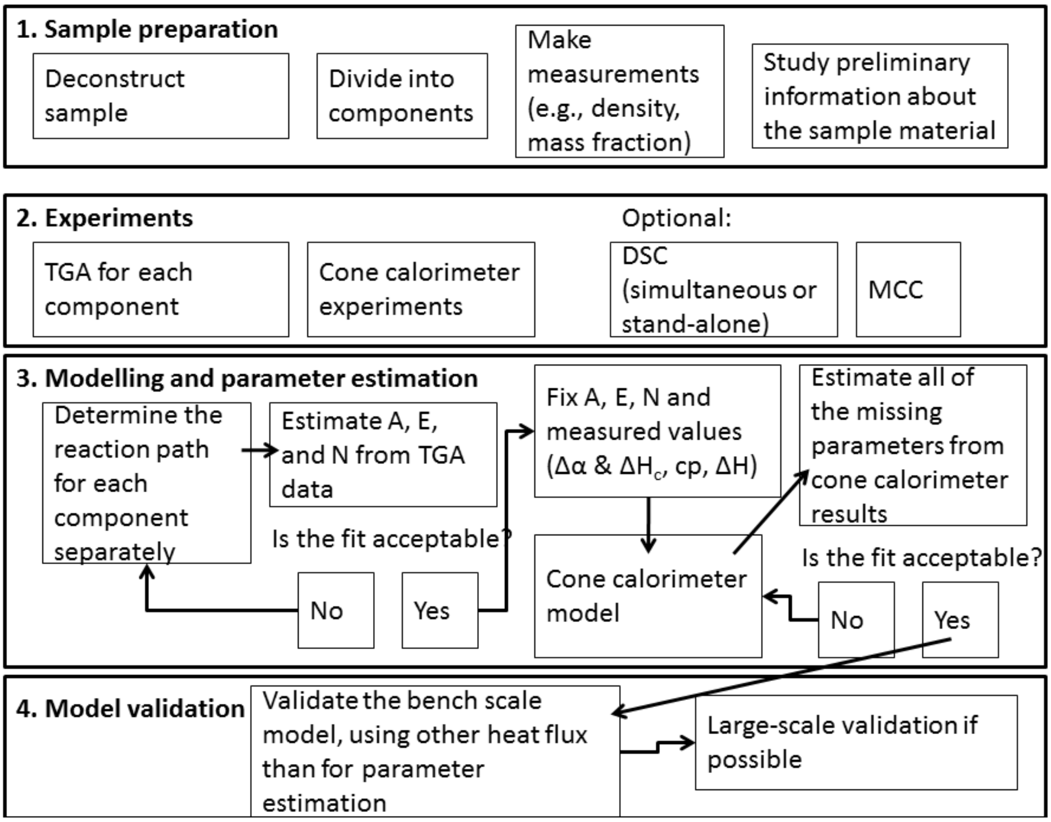


Figure 1. **Material modelling process and parameter estimation.**

Model validation is the Step 4 of the process. It is often neglected in hurry, but it is very important part of the modelling process. Validation means the assurance that the software / model meets the needs of the user, in this case, is able to predict the thermal degradation of a solid in different circumstances. In the simplest form, material model validation can be just comparing simulated and experimental cone calorimeter results at different heat fluxes than the parameter estimation has been made. However, often validation is understood to include larger scale experiments and simulations. The validation also gives an idea in which situations the model can be trusted to predict fire spread correctly, and in which case the results should be used with caution.

More information about the used experimental methods and models is available in [1].

2.2 Cable modelling

Cables are challenging to model due to their non-uniform structure and cylindrical geometry. There are basically two ways to model a cable in FDS: Using rectangular obstacles (OBST) or sub-grid-scale particles (PART).

OBST is the traditional way to model any sample in FDS, because it is very easy, fast and stable to calculate. The size of the OBST is limited to the grid cell size, and only Cartesian geometry is reasonable. The fire spread in a cable tray performs without problems, but the air flow between cables is not possible.

PART is a method under development. The idea is to describe a cable with a particle that has the same properties as a solid surface. As the particle would not be physically limited by the grid, also cylindrical geometries are possible. The boundary conditions for the particle are still determined according to which grid cell the particle is located. Therefore each cable can be modelled with a series of consecutive particles, one in each grid cell of the cable length. If the radiation conditions are not uniform for all the directions, a particle can be *splitted* (or divided) into several smaller particles that each sees radiation from one direction. The mass of the particle is scaled accordingly. This makes calculation slightly slower, since the pyrolysis needs to be solved for each particle separately. Other drawbacks are related to radiation interaction (particles that are located in the same grid cell do not see each other's radiation), drag (in real cable tray there is only little free area that is not correctly modelled by particles) and the heat transfer through the cable (now, the heat transfer is only solved from surface to the centre, but sometimes the heat may also go through the whole cable, or from the sides).

3. Results

3.1 Step 1: Sample preparation

Sample cable #701 is identified as GENERAL CABLE® BICC® BRAND SUBSTATION CONTROL CABLE 7/C #12AWG 600V from year 2006 [2]. It includes three main components: Sheath, insulation and conductor. The measured mass fraction and density of components are listed in Table 1.

Table 1. **Composition of cable #701.**

Component	**Material**	**Mass fraction**	**Density**
		(-)	(kg/m^3)
Sheath	PVC	0.24	1542
Insulation	PE	0.18	1153
Conductor	Copper	0.58	8954

PVC (Poly(vinyl Chloride)) is a thermoplastic polymer with wide range of applications. It is used both in rigid (C_2H_3Cl) and in plasticized ($C_{26}H_{39}O_2Cl$) form. In cables PVC is in plasticized form. Pure PVC degrades in two steps. First degradation reaction occurs between 200 and 300 °C, and releases hydrochloric acid (HCl). HCl is not combustible, but it is highly corrosive and therefore harmful for people and environment. The degradation of the remaining polymer starts immediately after the release of HCl, yielding small amounts of aromatics (mainly benzene). The second degradation step happens around 450 °C where combustible gas is released [7]. In the plasticized form of PVC, also a component called plasticizer is needed. It degrades almost simultaneously with the first reaction of PVC, releasing combustible gas. That makes plasticized PVC more flammable than its pure form [8]. Besides of nominal PVC and plasticizer, third significant component of a PVC cable material is the filler. Typically this is calcium carbonate ($CaCO_3$). It degrades thermally at the high temperatures, but is not combustible [9]. More detailed literature review on the thermal degradation of PVC and its additives can be found from [1].

3.2 Step 2: Experiments

TGA experiments for sheath and insulation components were performed at 10 K/min in nitrogen, and also in air for sheath. Conductor was not tested in TGA, because it does not degrade. Cone calorimeter experiments were performed at 25, 50 and 75 kW/m^2. The cone calorimeter samples consisted of seven parallel, 10 cm long cables wrapped in an aluminium foil. Additionally also MCC experiments were performed for sheath and insulation components at 60 K/min. DSC experiments were also performed at 10 K/min for sheath (both air and N_2) and for insulations (only N_2). The experimental results are shown in Figure 2. TGA, DSC and some MCC experiments were performed at VTT, while cone calorimeter and other MCC experiments were made at NIST [2]. From TGA results it can be seen that there are three main reaction steps in the thermal degradation of sheath and insulation materials (third one is only a small one). Compared with MCC results, it can be seen that the two first reactions release significant amounts of combustible gases for both components, and in the third reaction heat release is less important. The mass loss and heat release at each reaction step are listed in Table 2. In air the mass loss of sheath is greater than in N_2, which suggests that surface oxidation may occur at the high temperatures in the presence of air. The difference in mass loss due oxidation was 7.7 %. The MCC results at NIST and VTT were slightly different, and they both are included in the table. The cone calorimeter results show that there is some experimental variation between the replicated tests. This phenomenon is strongest at low heat fluxes. Variation is greater in the heat release rate measurement than in mass loss rate. The interpretation of DSC results is more complicated, because the baseline changes so much in the end of the experiments. It is almost impossible to define peaks and determine whether they are endo- or exothermic.

For validation, radiant panel experiments are used. Those results were reported in report of Christifire campaign [2].

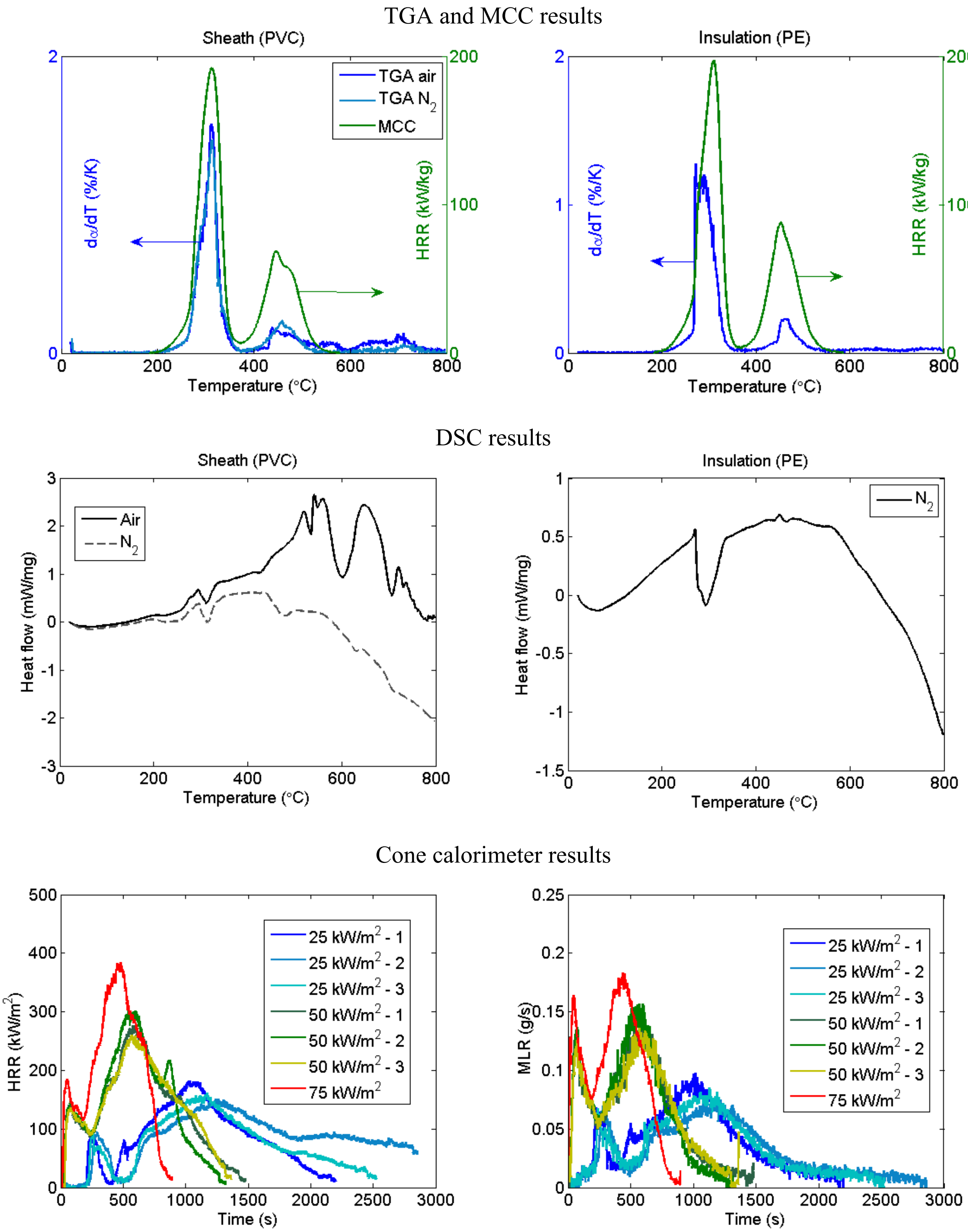


Figure 2. **Experimental small and bench scale results of Christifire cable #701.**

Table 2. **Numerical results of TGA and MCC experiments.**

Sheath			
	1	2	3
Δm (mass fraction)	0.6	0.137	0.04
Q - NIST (MJ/kg)	9.2	4.9	0
Q - VTT (MJ/kg)	9.7	5.1	0
Insulation			
Δm (mass fraction)	0.57	0.12	0.052
Q - NIST (MJ/kg)	8.2	4.8	0
Q - VTT (MJ/kg)	8.7	4.7	0

3.3 Step 3: Modelling and parameter estimation

The reaction path can be defined in many ways. The choice of the reaction path is not very significant for the overall fit of the cone calorimeter results, but it does have a great effect on the model parameters [1]. Therefore the decision is up to the modeller. Two main standpoints can be identified: A simple reaction path that only tries to repeat the correct mass and heat flow at each temperature, and a more sophisticated model that intents to understand the real thermal degradation of the sample material in some level. In the latter one, the information collected in the Step 1 can be used. Both methods are applied in this example.

Following the simplified way (Method 1), each reaction is assumed to release combustive gas with heat of combustion 46.45 MJ/kg, and one inert gas (water vapour). No residue is yielded until the last reaction step. The gas yields are calculated using the TGA and MCC results listed in Table 2, and they are shown in Table 3.

Table 3. **Inert (I) and fuel (F) yields in the reaction path of Method 1.**

		Sheath		**Insulation**	
		NIST	VTT	NIST	VTT
1	y_I	0.67	0.65	0.69	0.67
	y_F	0.33	0.35	0.31	0.33
2	y_I	0.23	0.20	0.14	0.15
	y_F	0.77	0.80	0.86	0.85
3	y_I	0.04	0.04	0.17	0.17
	y_F	0.00	0.00	0.00	0.00

In the more realistic reaction path, Method 2, it is assumed that PVC consists mainly of pure PVC, plasticizer and CaCO3, as discussed in Step 1. PE is similarly assumed to consist of pure PE, plasticizer and CaCO3. With some realistic boundary conditions, the reaction path and parameters are as shown in Table 4. According to the parameters, before any degradation sheath material includes 51.4 % pure PVC, 26.8 % plasticizer and 21.8 % CaCO3. Insulation includes 12 % pure PE, 57 % plasticizer and 31 % CaCO3. Of course, these values are just model parameters and may not be the real mass fractions of the components.

Table 4. **Reaction path and parameters of Method 2. Comp 1 is pure PVC, Comp 2 is plasticizer and Comp 3 is $CaCO_3$.**

		Sheath		Insulation	
		NIST	VTT	NIST	VTT
Comp 1-1	y_I	0.602	0.589	0.00	0.00
	y_F	0.043	0.028	1.00	1.00
	ΔH_c (MJ/kg)	49.1	49.7	39.7	39.3
	Residue	Comp 1-2	Comp 1-2	-	-
	m0 (mass fraction)	0.514	0.449	0.12	0.12
Comp 1-2	y_I	0.00	0.00	-	-
	y_F	0.751	0.797	-	-
	ΔH_c (MJ/kg)	35.8	37.4	-	-
	Residue	Char	Char	-	-
	m0 (mass fraction)	0.00	0.00	-	-
Comp 2-1	y_I	0.00	0.00	0.00	0.00
	y_F	1.00	1.00	1.00	1.00
	ΔH_c (MJ/kg)	30.2	28.0	14.4	15.3
	Residue	-	-	-	-
	m0 (mass fraction)	0.268	0.323	0.57	0.57
Comp 3-1	y_I	0.184	0.175	0.168	0.168
	y_F	0.00	0.00	0.00	0.00
	ΔH_c (MJ/kg)	0	0	0	0
	Residue	Char	Char	-	-
	m0 (mass fraction)	0.218	0.228	0.31	0.31

After determining the reaction path, kinetic parameters are estimated from the TGA results using genetic algorithm [1]. The parameters are listed in Table 5 and the comparison of experimental and fitted TGA results are shown in Figure 3.

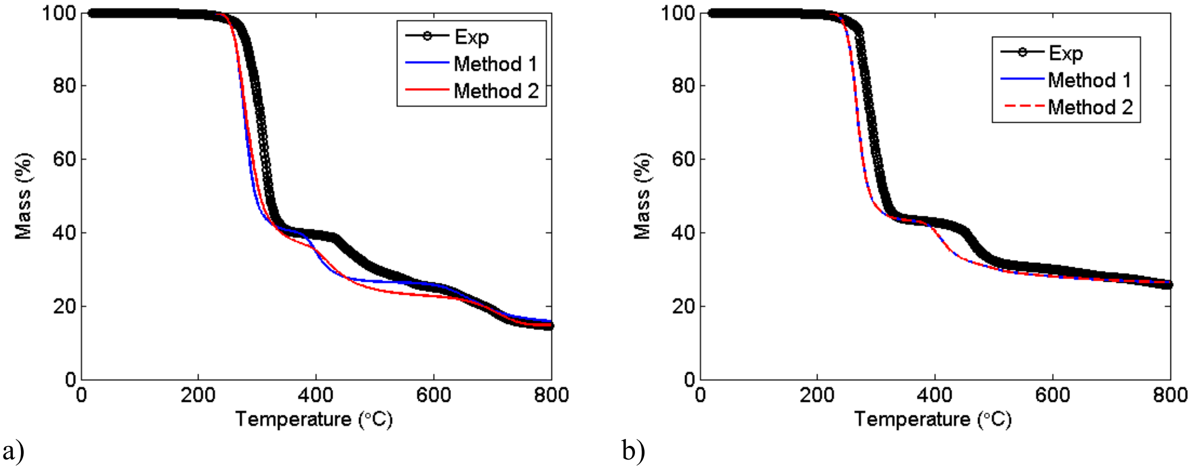


Figure 3. **Visual comparison of experimental and numerical TGA results for a) sheath and b) insulation component of cable #701 at 10 K/min in N_2.**

Table 5. **Kinetic parameters of sheath and insulation of cable #701.**

Material	Method	Component (i) Reaction (j)	A (s^{-1})	E (mol/kJ)	N
Sheath (PVC)	Method 1	i = j= 1	$3.6 \cdot 10^{21}$	$2.4 \cdot 10^{5}$	2.87
		i = j = 2	$1.2 \cdot 10^{29}$	$3.8 \cdot 10^{5}$	4.10
		i = j = 3	$5.1 \cdot 10^{21}$	$3.0 \cdot 10^{5}$	2.67
		i = j = 4	$6.0 \cdot 10^{12}$	$2.5 \cdot 10^{5}$	1.4
	Method 2	i = 1,2, j = 1	$2.1 \cdot 10^{26}$	$2.8 \cdot 10^{5}$	3.69
		i = 1, j = 2	$2.0 \cdot 10^{25}$	$3.2 \cdot 10^{5}$	4.91
		i = 3, j = 3	$9.8 \cdot 10^{24}$	$2.9 \cdot 10^{5}$	0.96
		i = 1,3 j = 4	$2.5 \cdot 10^{10}$	$2.2 \cdot 10^{5}$	1.0
Insulation (PE)	Both methods	i = j = 1	$1.26 \cdot 10^{25}$	$2.7 \cdot 10^{5}$	3.20
		i = j = 2	$1.9 \cdot 10^{27}$	$3.6 \cdot 10^{5}$	3.70
		i = j =3	$1.6 \cdot 10^{12}$	$2.1 \cdot 10^{5}$	4.41

As the fit is evaluated as accepted, the next step is to fix the earlier determined kinetic parameters, heat of combustions and densities, and make a model for cone calorimeter. In this example only OBST model is presented. It consists of five, rectangular layers:

1. Sheath 2.1 mm
2. Insulation 2.1 mm
3. Conductor 1.7 mm

4. Insulation 2.1 mm
5. Sheath 2.1 mm
6. (Insulating backing) 20 mm.

The remaining parameters (k, cp, ΔH, ε and residue density ρ) are estimated from cone calorimeter results at 50 kW/m2. The results are shown in Figure 4 and the parameter values listed in Table 6. The fit with both methods is equally good, but the model parameters are completely different. As the fit is found acceptable, next step is model validation.

Table 6. **Thermal parameters for cable #701.**

			Method 1	Method 2		
				Comp 1 (Polymer)	Comp 2 (Plasticizer)	Comp 3 ($CaCO_3$)
Sheath	Reaction 1	k (W/(mK))	0.147	0.146	0.185	0.48
		c_p (kJ/(kgK))	3.22	3.4	2.8	3.5
		ΔH (kJ/kg)	1607	206	1112	1669
		ε	0.7	1.0	1.0	1.0
	Reaction 2	k (W/(mK))	0.175	0.2	-	-
		c_p (kJ/(kgK))	3.45	2.26	-	-
		ΔH (kJ/kg)	1425	1783	-	-
		ε	1.0	1.0	-	-
	Reaction 3	k (W/(mK))	0.103	-	-	-
		c_p (kJ/(kgK))	3.5	-	-	-
		ΔH (kJ/kg)	43	-	-	-
		ε	1.0	-	-	-
	Reaction 4	k (W/(mK))	0.1	0.2	-	0.2
		c_p (kJ/(kgK))	3.5	2.5	-	2.5
		ΔH (kJ/kg)	40	1500	-	1500
		ε	1.0	1.0	-	1.0
	Residue	ρ (kg/m^3)	344	70	-	274
		k (W/(mK))	0.122	0.188	-	0.188
		c_p (kJ/(kgK))	3.5	2.0	-	2.0
		ε	0.85	1.0	-	1.0
Insulation	Reaction 1	k (W/(mK))	0.783	0.246	-	-
		c_p (kJ/(kgK))	3.36	1.9	-	-

		ΔH (kJ/kg)	1408	1760	-	-
		ε	1.0	1.0	-	-
	Reaction 2	k (W/(mK))	1.0	-	0.59	-
		c_p (kJ/(kgK))	3.4	-	3.0	-
		ΔH (kJ/kg)	1516	-	691	-
		ε	1.0	-	1.0	-
	Reaction 3	k (W/(mK))	0.087	-	-	0.285
		c_p (kJ/(kgK))	2.74	-	-	2.9
		ΔH (kJ/kg)	445	-	-	353
		ε	1.0	-	-	1.0
	Residue		297	-	-	297
		k (W/(mK))	0.01	-	-	0.338
		c_p (kJ/(kgK))	1.29	-	-	1.29
		ε	1.0	-	-	1.0

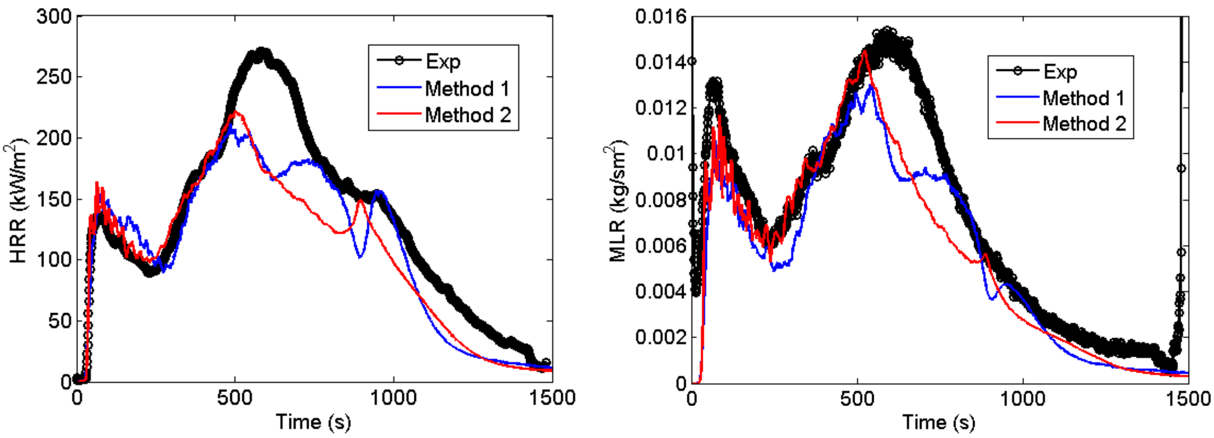


Figure 4. **Comparison of experimental and numerical cone calorimeter results at 50 kW/m^2.**

3.4 Step 4: Model validation

The simplest form of model validation is to compare experimental and simulated cone calorimeter results at different heat fluxes. This confirms that the parameters are generally able to predict heat release and mass loss rate in different conditions and the parameters have not been "over-fitted" to give good results just at one heat flux. Here, the results were tested at 25 and 75 kW/m^2. The results are shown in Figure 5. The accuracy of the prediction is better at 75 kW/m^2, but it must be remembered that also experimentally there was larger variation in the results at low heat fluxes.

The larger scale validation is done using the experimental results of radiative panel. The larger the experimental method, the larger is also the variation between the experimental results. Therefore the comparison is done at several nominal heat fluxes and test configuration (more details in [2]). The model of Method 1 is chosen for this demonstration, since it was observed that there are no significant differences between results of different models. The results are seen in Figure 6. It can be seen that in some cases (especially RP-3) the model prediction is very good especially in the beginning of the experiment. In some other cases (RP-30), the prediction is significantly worse, although experiment RP-30 also has very long ignition time compared to any other test.

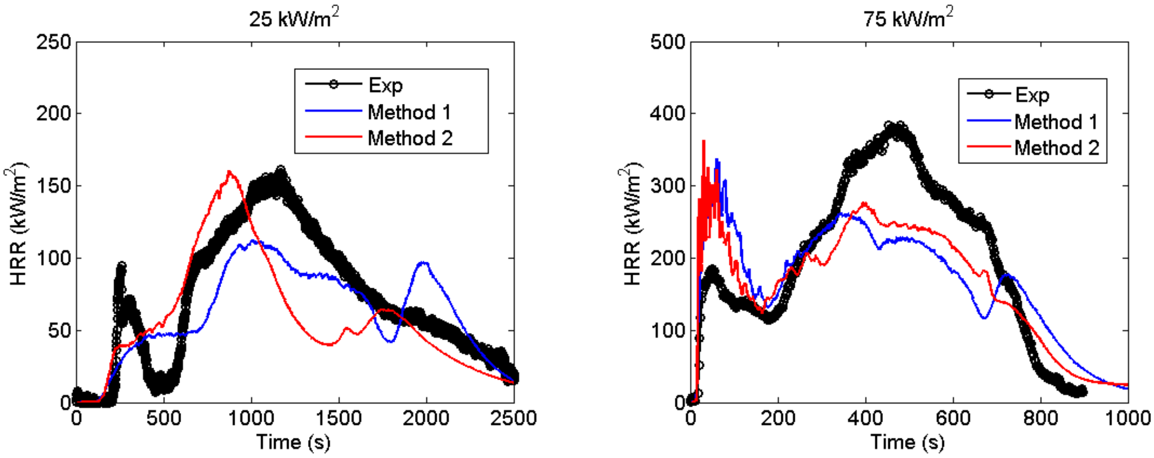


Figure 5. **Validation of cone calorimeter models at 25 and 75 kW/m^2.**

High heat flux **Low heat flux**

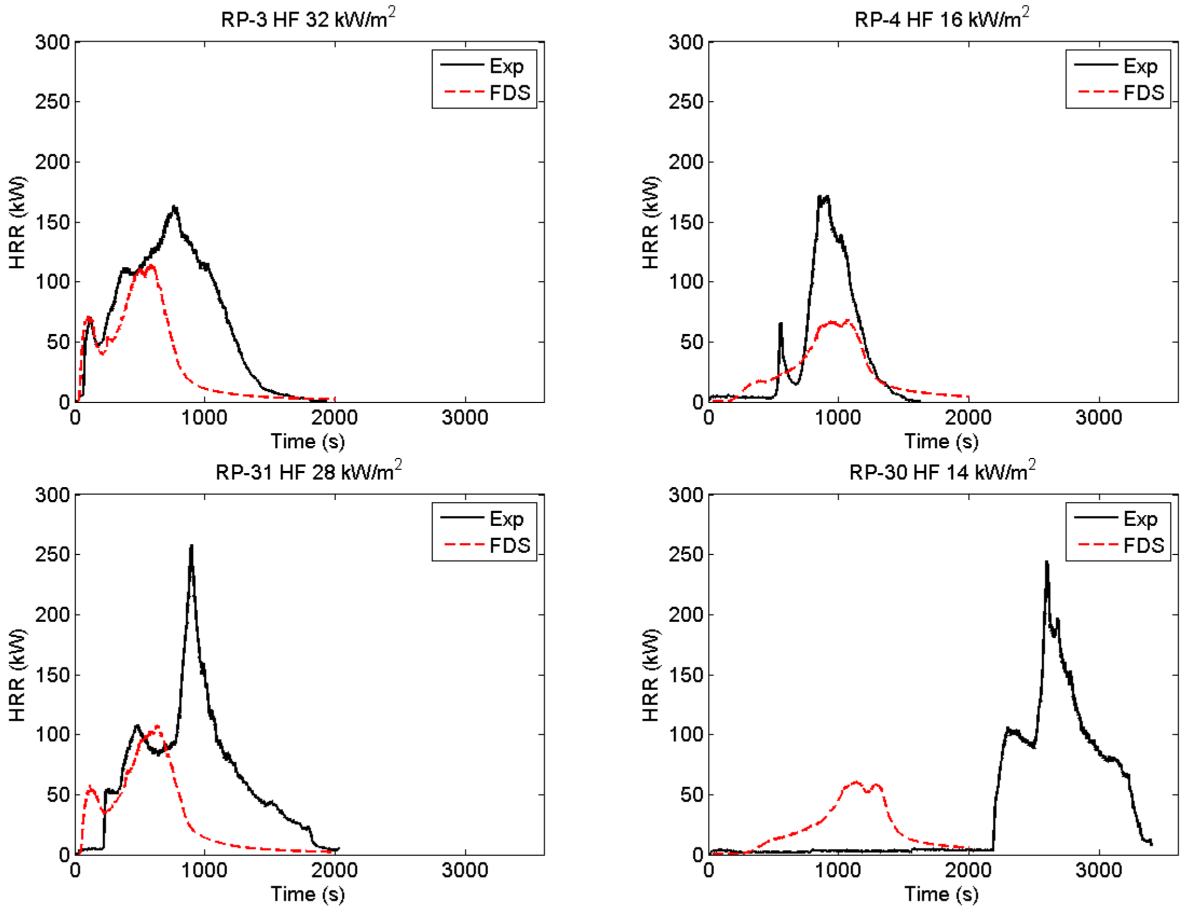


Figure 6. **Results of radiative panel simulations. Number of cables is 44 in all except RP-4 it is 22. In the upper row the cable packing is loose, while in the lower row it is dense.**

4. Conclusions

The pyrolysis modelling process was presented and demonstrated step-by-step using Christifire cable #701. Two alternative models were made for the cable in the cone calorimeter scale. It was noticed that the model fit was equally good for both models, but the used model parameters were significantly different. Therefore one must remember not to mix reaction paths and parameters from different sources.

The models were validated in bench-scale using cone calorimeter results at different heat fluxes. The prediction ability was better at higher heat fluxes. The experimental variation of the results was also higher at lower heat flux, so the result is logical. The larger scale validation was made using radiative panel results. The accuracy of the predicted heat release rate was again better at higher heat fluxes.

The process of pyrolysis modelling is a complicated process with several steps, choices and decisions. For accurate models, one must follow all the steps and be able to justify the decisions made. The modelling process was demonstrated in this paper carefully addressing all the phases of the process, at the same time describing the modelling of a PVC cable from small scale to large scale.

Acknowledgements

Authors would like to thank Dr. Kevin McGrattan from NIST for the experimental results of Christifire campaign. Dr. Tuula Leskelä from Aalto University School of Chemical Technology is also acknowledged for the TGA and DSC results. The work was partially funded by The State Nuclear Waste Management Fund (VYR).

References

[1] Matala, A. Methods and applications of pyrolysis modelling for polymeric materials. VTT Science 44. Doctoral thesis, 2013.

[2] McGrattan, K., Lock, A., Marsh, N., Nyden, M., Bareham, S., Price, M., Morgan, A.B., Galaska, M., Schenck, K. Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), Phase 1: Horizontal Trays (NUREG/CR-7010, Volume 1). 2012.

[3] Tunturivuori, L. Updating of the fire PRA of the Olkiluoto NPP units 1 and 2. PSAM-11 – ESREL 2012. 25-26 June, Helsinki, 2012.

[4] McGrattan et al. Fire Dynamics Simulator. User's Guide. NIST Special Publication 1019 (2012).

[5] Matala, A., Hostikka, S. Pyrolysis modelling of PVC cable materials. Fire safety science 10: 917-930. 2011.

[6] Lyon, R.E., Safronava, N., Oztekin, E. A simple method for determining kinetic parameters for materials in fire models. Fire Safety Science 10: 765-777, 2011.

[7] Papazoglou, E.S. Flame retardants for plastics. Chapter 4 in Handbook of building materials for fire protection (Harper, C.A. edit) McGraw.Hill Handbooks, New York, 2004.

[8] Marcilla, A., Beltrán, M. Effect of the plasticizer concentration and heating rate on the thermal decomposition behaviour of PVC plastisols. Kinetic analysis. Polymer degradation and Stability 60 (1): 1-10, 1998.

[9] Wypych, G. PVC Formulary. ChemTec Publishing, 2009.

Insights on Recent Regulatory Guideline of Fire Protection in Finland

Matti Lehto, Pekka Välikangas
Radiation and Nuclear Safety Authority (STUK), Finland

Abstract

Radiation and Nuclear Safety Authority (STUK) issues detailed regulations that apply to the safe use of nuclear energy and to physical protection, emergency preparedness and safeguards. STUK upgraded the nuclear safety regulation in December 2013. According to defence-in-depth, fires have to be prevented in all stages from ignition to spreading across fire compartments. Consequences have to be minimized in order to ensure necessary safety functions during and after fires. Final goal is to ensure fire compartments. Technical design requirements for fire protection were not changed, but more guidelines are stated in order to ensure systematic design and safety goals.

1. Introduction

Radiation and Nuclear Safety Authority (STUK) issues detailed regulations that apply to the safe use of nuclear energy and to physical protection, emergency preparedness and safeguards. STUK revised main part of the nuclear safety regulation, known as YVL guides, in December 2013 (the Guides are published in Finnish; English translations are under preparation). Fire protection guide is one of the YVL guides and the previous guide YVL 4.3 was replaced with the new guide coded as YVL B.8 “Fire Protection at a Nuclear Facility”. The most significant new issues in the Guide B.8 are guidelines for different licensing phases and for defense-in-depth (DiD) principles. All requirements are individually coded in the revised Guides.

Fukushima nuclear accident launched large global assessment of nuclear safety. Nuclear safety is under verification for conditions, where external natural threats are suggested to overlap existing design basis requirements. STUK’s YVL Guides include additional requirements based on lessons learnt from Fukushima accident and other recent incidents.

The task of the Radiation and Nuclear Safety Authority (STUK) as the national authority responsible for oversight of the safety of the use of nuclear energy is based on the Nuclear Energy Act (990/1987) [1] and the Nuclear Energy Decree (161/1988) [2]. STUK’s oversight includes the oversight of the fire protection arrangements of nuclear facilities in so far as they affect the nuclear and radiation safety of the facilities.

When YVL Guides set requirements for nuclear facilities, reference is made, under the Nuclear Energy Act (990/1987) [1], to facilities necessary for producing nuclear energy (nuclear power plants), including research reactors, facilities performing extensive final disposal of nuclear wastes, and facilities used for extensive fabrication, production, use, handling, storage of nuclear materials or nuclear wastes. Requirements for nuclear facilities always apply to nuclear power plants unless a requirement separately says they only apply to other nuclear facilities.

STUK's Guide YVL B.8 applies to the planning and implementation of fire protection during the design, construction and operation of the nuclear facility. The Guide shall be applied to the decommissioning of nuclear facilities. This guide shall be complied with at the entire plant site and in all its buildings. As regards fire protection at a nuclear facility construction site, this Guide shall apply whenever fire protection is significant for the safety of nearby nuclear facilities and to ensure fulfilment of the design criteria of the nuclear facility under construction.

The Government Decree (717/2013) [3] presents requirements for the safety design of nuclear power plants:

- Section 12 requires implementation of the defence in depth principles to prevent accidents and to mitigate their consequences.
- Section 18 requires that the internal events to be considered include at least fire, floods, explosions and component failures.
- Section 19 presents requirements for the nuclear power plant's control room arrangements.
- Sections 21-26 present requirements for the nuclear power plant's construction, commissioning, operation, processing of operational experiences, safety research and the Operational Limits and Conditions.
- Sections 28-30 present requirements for the organisation and personnel of a nuclear power plant.

The Government Decree (736/2008) [4] presents requirements for the safety design of the final disposal of nuclear waste:

- Sections 17 and 18 present requirements for the construction, commissioning and operation of a nuclear facility.
- To prevent operational occurrences and accidents, Section 8 requires, among other things, that in a nuclear waste facility, the placement and protection of systems alongside operative methods shall ensure that fires, explosions or other events inside the facility do not pose a threat to safety.

The Ministry of the Environment issues technical regulations and guidelines on construction and structural fire protection [5]. The building inspection authority in each municipality sees to it that the regulations and guidelines issued by the Ministry are complied with in all construction activities.

Leadership and control of fire and rescue services, as well as the availability and quality of its services, rests with the Ministry of the Interior. The Ministry is also responsible for the preparation and arrangement of fire and rescue services at national level and for co-ordination of the performance of different ministries involved in the fire and rescue services under the Rescue Act (379/2011) [6] and the Government Decree (407/2011) on fire and rescue services [7]. Regional State Administrative Agencies are responsible for the duties of rescue services in their sphere of activity. Municipalities are responsible in co-operation for fire and rescue services in a region determined by the Government (regional fire and rescue services). As regards the requirements, design, installation, maintenance, inspection and demonstration of conformity of the equipment of the rescue services, the Rescue Equipment Act (10/2007) [8] shall be observed.

The Government Decree (917/1996) [9] and the Ministry of Trade and Industry Decision (918/1996) [10] present the requirements for equipment and protective systems intended for potentially explosive atmospheres. The Government Decree (576/2003) [11] presents the requirements for prevention of personnel hazards caused by potentially explosive atmospheres.

Finnish Safety and Chemicals Agency (Tukes) and the Ministry of Social Affairs and Health provide guidelines on the application of the ATEX legislation in Finland [12].

STUK's activities do not affect any oversight activities required in the Land Use and Building Act (132/1999) [13], the Land Use and Building Decree (895/1999) [14], the Rescue Act (379/2011) [6] and the Government Decree (407/2011) on Rescue Services [7], unless otherwise agreed between the authorities.

STUK's Guide YVL B.8 describes fire protection inspections performed by STUK during the design, construction and operation of the nuclear facility. Furthermore, it presents the requirements for fire protection documents to be submitted to STUK.

In addition to the fire protection requirements of the Guide YVL B.8, the following STUK's Guides also contain fire protection related requirements to be followed:

- Guide YVL A.1 (Regulatory oversight of safety in the use of nuclear energy) sets forth requirements for nuclear facility design and oversight.
- Guide YVL A.3 (Management system of a nuclear facility) sets forth detailed requirements related to the management system and quality management.
- Guide YVL A.5 (Construction and commissioning of a nuclear facility) sets forth requirements for the management and oversight of the construction project at different stages of a nuclear facility's construction.
- Guide YVL A.6 (Conduct of operations at a nuclear power plant) sets forth requirements for the operation of a nuclear power plant, such as for outages.
- Guide YVL A.7 (Probabilistic risk assessment and risk management of a nuclear power plant) sets forth requirements for probabilistic fire risk assessments.
- Guide YVL A.11 (Security of a nuclear facility) sets forth requirements for physical protection at a nuclear facility and its planning.
- Guide YVL B.1 (Safety design of a nuclear power plant) sets forth requirements for the nuclear power plant's safety design and the design of systems important to safety.
- Guide YVL B.7 (Provisions for internal and external hazards at a nuclear facility) sets forth requirements for nuclear facility layout design and the design to protect against internal and external threats.
- Guide YVL E.6 (Buildings and structures of a nuclear facility) sets forth requirements for the design of civil structures.
- Guide YVL E.7 (Electrical and I&C equipment of a nuclear facility) sets forth electrical equipment specific requirements for protection against fire load induced explosions.

2. Design Requirements

2.1 General Requirements

Under Section 18 of the Government Decree 717/2013 [3], structures, systems and components important to safety of a nuclear power plant shall be designed and located as well as protected in a way to make the likelihood of internal events (such as fires) small and their effect on facility safety insignificant.

The fire protection for the nuclear facility shall be so planned that during and after a potential fire situation the nuclear facility can be brought to a safe state and the release of radioactive substances into the environment can be prevented.

The licensee can propose that also foreign regulations and guides be applied in designing the nuclear facility's fire protection arrangements. It shall then be demonstrated, however, that they form a feasible entity. The application of foreign regulations and guides is subject to STUK's approval.

For the inclusion of all aspects of fire protection, an expert responsible for fire protection design shall be nominated for the duration of the nuclear facility's design and construction. The expert shall have sufficient qualifications and experience in nuclear, radiation and fire safety. Management of the entirety of the nuclear facility's fire protection arrangements places specific requirements on the combination of several design areas, such as facility layout, structural, heating/ventilation/air-conditioning, as well as electrical and I&C design.

In addition to the design requirements of the Guide YVL B.8, the following shall be complied with in the design of nuclear facilities:

- The fire and building legislation in force in Finland.
- For applicable parts, the practices of risk-informed fire protection design for nuclear power plants described in the IAEA Guides [15 ... 21] as well as in a technical report [22].
- The practices of the WENRA [23].

2.2 Defence in depth

The nuclear facility's fire protection shall be based on the defence in depth principle, which aims to:

- Prevent the outbreak of a fire.
- Rapidly detect and extinguish ignited fires.
- Prevent fire growth and spreading of a fire.
- Contain a fire so that the facility's safety functions can be reliably performed irrespective of the effects of the fire.

Advanced and reliable technical designs and methods shall be used to prevent fire ignition, including e.g.:

- Minimisation of the danger of ignition by the use of construction materials allowable, in accordance with the National Building Code of Finland [5].
- Protection and monitoring of equipment causing the risk of a fire, e.g. monitoring of vibration and oil leaks of rotating apparatuses (turbine generators, diesel generators and large pumps), transformer hydrogen analysers and electric arc protections of switchgears.
- Ensuring fire protection relating to temporary fire loads and fire hazardous components, supervision of work and administrative procedures as well as work related personnel training.

A fire shall be detected and promptly extinguished by active fire prevention arrangements including e.g.:

- Automatic fire detection system covering the entire facility.

- Protection of components containing significant fire hazards by fixed extinguishing systems.
- Fire protection during fire hazardous work.
- Operative fire fighting.

Fire growth and spread shall be prevented and the effects mitigated by reliable technical means including:

- Fire separation of buildings and safety divisions.
- Fire compartmentation and local fire protection.
- Stopping or rerouting of ventilation to restrict the supply of oxygen and prevent smoke spread.
- Extraction of smoke and combustion gases.

Implementation of the concept of defence in depth in fire protection shall be assessed by analyses, which focus at least on the following:

- Rooms where the fire separation of safety divisions cannot be implemented by means of a fire wall according to standards.
- The containment, annulus and control room as well as areas where the zone affected by a design basis fire is smaller than the entire fire compartment.
- Rooms where the fire load contributing to a fire is, in a design basis fire, assumed to be smaller than the fire load of the entire fire compartment or a single component (e.g. the fire compartment may contain large cable concentrations, a large transformer and oil systems that do not inherently burn completely due to layout and/or structural protection).

2.3 Structural Fire Protection

General:

The nuclear facility shall be designed in such a way that structural fire protection together with the facility´s functional design and layout design ensure the safety of the facility during fire situations as far as possible without active fire fighting operations.

Incombustible construction materials or materials with extremely limited combustion shall be used in structural elements.

Fire Resistance Classes and Separation of Buildings:

Buildings are divided into three fire classes in part E1 of the National Building Code of Finland (RakMK) [5]. Buildings containing systems important to the nuclear power plant’s safety shall be designed as Class P1 buildings. The fire class of buildings containing systems other than those important to safety is determined according to the regulations and guidelines of parts E1 and E2 of the RakMK.

The minimum fire requirement for the outer walls and roof of safety classified buildings is the fire resistance class EI-M 120 of RakMK part E1. If two buildings are conjoined, they shall be separated by a fire wall that complies with the fire resistance requirements of RakMK part E1 and has a minimum fire resistance class of EI-M 120.

The load-bearing structures of the buildings of nuclear power plants shall be constructed in compliance with the regulations of RakMK part E1 in accordance with the fire resistance class and fire load category of the building. Load-bearing structures shall at least meet the fire resistance rating R 60. The fire resistance rating (R) of a fire compartment's load-bearing structures shall, however, be at least equal to the fire resistance rating of the walls enclosing the fire compartment, in terms of fire insulation (I) and integrity (E).

Fire Separation of Safety Divisions:

STUK's Guide B.8 sets forth the requirements for the fire separation of safety divisions. Additional requirements for separation between safety divisions are given in STUK's Guides YVL B.1 and YVL B.7.

Safety divisions shall be separated by structures having a fire resistance rating of at least EI-M 120. If the safety division separation requirement of EI-M 120 is inadequate due to heavy fire loads, the rating of the structures shall fulfil the requirements accordant with fire loads, or their fire resistance rating shall be justified by fire hazard analyses.

In the separating structural elements between safety divisions, any elements reducing fire safety, such as doors, hatches and penetrations for ventilation, pipes and cables, shall be avoided as much as possible. In case these must be installed in structural elements between safety divisions, they shall fulfil the same fire resistance class requirement as the separating structural element.

Doors and hatches between safety divisions shall be kept locked during normal operation of the plant and they shall be equipped with continuous position monitoring. Separating fire doors shall be self-closing and self-bolting.

Fire Compartmentation:

Fire compartmentation shall be based on compartmentation by storey and compartmentation by use. Rooms with varying purposes of use, such as control rooms, computer rooms, electrical and switchgear rooms, cable rooms, battery rooms and active carbon filter rooms, shall be separated as their own fire compartments.

Heavy fire load concentrations or compartments where the risk of fire is high shall be separated into individual fire compartments. The amount of combustibles shall be minimised in areas and rooms important to safety.

The separating structural elements of compartments shall fulfil the fire resistance class requirements of the RakMK part E1 [5]. The minimum fire resistance class shall be EI 60.

The fire resistance rating of doors and hatches in separating structural elements other than those between safety divisions shall be at least half of that required for the structural element (wall, floor or roof):

- The fire resistance class of separating doors and hatches shall be at least equal to EI 60.
- Separating fire doors shall be self-closing and self-bolting.

Dampers as well as cable, ventilation and piping penetrations shall fulfil the integrity and insulation requirements (EI) for the penetrated separating structural element.

Protection Against Fire Load Induced Explosions:

Explosions and electric arcs as well as their consequent effects such as missiles shall be taken into account in designing fire protection arrangements at nuclear power plants. Protection shall be provided against explosions occurring in consequence of fires.

The nuclear power plant's design shall provide protection against the risk of explosions and high energy electric arcs in accordance with the defence in depth principle:

- To prevent explosions and arcs by monitoring and protection systems.
- To minimise the risk for plant safety from explosions and arcs.
- To limit the spread of the effects of an explosion and arc.

Combustible liquids or gases, which are not part of the facility's processes and could cause explosions, shall not be permanently or temporarily located in rooms important to plant safety, or in their immediate vicinity. The design of the facility and its fire protection shall take into account the spread of gases, gas mixtures and liquids far from the leak point before they ignite or explode.

The generation of conditions prone to explosions and high energy electric arcs shall be primarily prevented by means of design solutions (e.g. in tanks, piping and electrical rooms important to safety, such as switchgear rooms and battery rooms).

Pressure relief along controlled routes (e.g. pressure relief hatches of rooms) to prevent structural failures and collapse of rooms/buildings involving the risk of explosion shall be ensured in the design of rooms and buildings.

The possibility of high energy electric arcing shall be taken into account in the design of rooms containing electrical equipment and in the choice of the equipment:

- Switchgear cabinets important to safety shall be provided with protection against electric arc, which limit the duration of arcs and the amount of total energy generated and released.
- Design shall consider the possibility of smoke causing an arc flash in the switchgear.

In addition to fires, possibility of a high energy electric arc or a rapid, explosive energy discharge shall be taken into account in transformer positioning and protection:

- During a high energy discharge, the rapid release of gas as well as the mixing and expansion of air and gas could cause a powerful fire and explosion.
- Large oil-cooled transformers shall be equipped with monitoring and protection systems (hydrogen monitors, gas relays) to prevent fires and electric arc flashes.
- Transformers containing large amounts of oil shall be placed sufficiently far from buildings and protected with structures and fire extinguishing systems.

Rooms shall be provided with adequate ventilation if the risk of explosive concentrations of gas or dust exists:

- Hydrogen build-up shall be considered in the design of battery room ventilation.
- The risk of a fire and an explosion of dust or gas mixtures in ventilation ducts shall be considered.

Hydrogen stations (e.g. needed for generator cooling) shall be located sufficiently far from buildings important to safety and their design shall consider explosion pressure waves. Other gas cylinders shall be located and stored in rooms specially designed for them.

Process systems containing combustible gas mixtures (e.g. the off-gas system) shall be placed far from safety divisions. Provision shall be made for filter fires and hydrogen fires with regard to potential explosions as well.

Containment and Annulus:

Safety divisions (redundant subsystems) inside the containment and in the annulus shall be housed in separate fire compartments whenever possible. Whenever the fire compartmentation between safety divisions is not possible inside the containment of the nuclear power plant, the operability of components important to safety as well as redundant subsystems shall be ensured by protective structures, separation by distance, fire resistant materials and fire insulation. The design concepts shall be assessed in accordance with the defence in depth principle utilising a risk-informed approach and also taking into account the aircraft crash resistance requirements of STUK's Guide YVL A.11.

The amount of fire load inside the containment shall be minimised. Safety system equipment including cables and impulse lines shall be so located and protected that the effects of a potential fire are limited to one safety division only.

Protection and fire protection of the lubrication oil system of the primary circulation pump/motor shall be designed in accordance with the defence in depth principle. Potential oil fires shall not endanger the facility's safety functions. Provision shall be made for oil leaks by means of oil collection and drainage systems whereby leaked oil is extracted to sealed collection tanks (fire shall be suppressed in the tanks).

Control room and Emergency Control Room:

Control rooms shall be placed in plant site locations safe from fire risks. Guidelines and requirements for control rooms are given in STUK's Guides YVL B.1 and YVL A.11.

Separation of the control room and the emergency control room from the rest of the facility and from each other shall be implemented in compliance with the requirements set for the separation of safety divisions. The control room and the emergency control room shall be their own fire compartments in accordance with RakMK part E1 [5], however their fire resistance shall be not less than EI-M120. The control rooms shall have separate ventilation systems whose structural separation is equivalent to that between safety divisions.

Control systems in the emergency control room shall be separated from the control systems of the control room and made into separate fire compartments in such a way that loss of equipment in the control room, or in any single fire compartment, does not prevent the functioning of controls in both the control room and the emergency control room. A corresponding requirement applies to emergency control posts outside the control rooms, which complement the vital functions of the emergency control rooms.

Cables important to safety that run from different safety divisions to the control room shall be routed through separate fire compartments. In case cables from different redundant systems must exceptionally be located in the same fire compartment (e.g. the cable space under the control desk):

- The cables shall be separated inside the compartment by means of distance, fire resistant materials and fire insulation.
- The fire compartment shall be equipped with effective and reliable fire detection systems and fire extinguishing systems.

The control room and the emergency control room shall be provided with overpressure ventilation to prevent smoke from entering the control room or the emergency control room in case of a fire located outside the room. Overpressure ventilation of the emergency control room can be replaced by locating the supply air centre of the control room and the emergency control room in such a way that their independence as regards smoke risk is reliably ensured. Overpressure ventilation shall be separate from other ventilation systems.

In case of a fire situation in a control room, the control room personnel shall be able to quickly and safely move from the control room to the emergency control room.

Access and Escape Rooms:

The nuclear facility shall feature an adequate number of appropriate, sufficiently spacious and easy-to-use access routes to enable safe exit from the facility. Access and escape route design shall comply with the regulations of RakMK part E1 [5].

The fire brigade shall be able to operate effectively at the plant during a fire situation. The design of attack routes for fire brigades shall be in compliance with the regulations of RakMK part E1.

The personnel shall be able to move within the plant to ensure the necessary safety functions during a fire or other accident. Emergency response operations shall be ensured by appropriate training.

Nuclear security shall be taken into account in the design of access and escape routes. Nuclear security is addressed in STUK's Guide YVL A.11. Requirements imposed by passage within the plant site and transport are given in STUK's Guide YVL B.7.

2.4Active Fire Protection

Automatic Fire Detection Systems:

To detect and locate a fire as quickly as possible, the nuclear facility buildings shall have extensive, sufficiently effective and reliable automatic fire detection systems. They shall be so designed that the location of a fire can be identified at least to any individual room. In large rooms containing systems important to safety it shall be possible to identify the location of the alarm with sufficient accuracy, even to a single detector within the room, if necessary.

The alarms of fire detection systems shall always be relayed to the facility unit's control room and to the plant fire brigade.

The selection and placement of fire detection equipment shall take into account the characteristic features of the compartment including ambient conditions, fire loads, ventilation and the significance of the compartment to the safety of the facility. If necessary, the fire detection systems can be supplemented with other appropriate monitoring systems.

Fire Extinguishing Water Systems and Fire Extinguishing Systems:

There may be several nuclear power plants at the site as well as other nuclear facilities (e.g. an interim storage for spent nuclear fuel, nuclear waste processing utilities and storages). If the on-site fire extinguishing water system serves several nuclear facilities, its capacity and significance in terms of safety during events threatening the entire facility site shall be assessed.

The nuclear power plant and other nuclear facilities at the site shall be equipped with fire water tanks, a fire water pumping station and fire water mains. Fire water volumes and the capacities of the fire water pumping stations shall be designed in accordance with sprinkler rules to supply water to the most extensive area requiring protection and taking into account potential fire spread. Furthermore, an adequate amount of fire water must be available for operative use by fire brigades. Requirements concerning fire extinguishing systems are set forth in the Ministry of the Interior Decree SM-1999-967/Tu-33 [24] on automatic fire extinguishing equipment. Guidelines regarding fire extinguishing systems are provided in standards [25–29].

To facilitate fast suppression of fires and to minimise damage and hazards, the nuclear power plant and other nuclear facilities shall be equipped with a fire water system and effective and reliable fire extinguishing systems. Location of the facilities, structural fire protection solutions and the amount of fire loads shall be taken into account in the design of the fire extinguishing systems of different fire compartments.

Irrespective of the layout design of the nuclear power plant or the amount of existing fire loads, at least the following rooms and systems shall be provided with fixed, sufficiently reliable and, if necessary, automatic fire extinguishing systems:

- Cable spreading rooms where compartmentation between safety divisions (redundant subsystems important to safety) is not realized.
- Cable rooms containing large cable concentrations with a fire load density of > 1200 MJ/m², unless it can be demonstrated that the development of a continuous cable fire in them is highly unlikely (defence in depth principle of fire protection shall be involved).
- Rooms and systems containing radioactive substances from which considerable amounts of radioactive substances can be released into rooms or the environment due to a fire, unless the risk is otherwise demonstrated to be insignificant.
- Where necessary, the components featuring heavy fire load (e.g. diesel generators, large transformers and other systems containing large amounts of oil).

The removal of fire water shall be arranged from rooms equipped with fixed water extinguishing systems or from rooms where large quantities of fire water are presumably needed in a fire situation. The effects of extinguishing water induced flooding shall be taken into account in the design and placement of these rooms. Loose parts shall also be taken into account in the removal of fire water.

The fire water systems of the nuclear power plant and other nuclear facilities shall be implemented in such a way that in case of the potential failure of a system part, the leak point can be isolated to limit the inoperative section of fire water system to the vicinity of the failure point.

STUK's Guide YVL B.7 prescribes that fire protection systems shall be so designed that their breaking or inadvertent operation does not significantly reduce the capability of structures, systems and components important to safety to carry out their safety functions.

The seismic resistance of fire water and extinguishing systems is verified in accordance with STUK's Guide YVL B.7. Systems and components to be protected shall be determined by risk-informed assessment in accordance with STUK's Guide YVL B.2. This applies to extinguishing water tanks, pumping stations, piping, and protection against pipe breaks in particular.

Operative Fire Fighting:

The nuclear power plant shall have an operative fire fighting readiness consisting of fire protection performed by the plant fire brigade, plant personnel and off-site fire brigades. This includes the on-site movable fire fighting equipment.

On the nuclear power plant site or in its immediate vicinity there shall be a plant fire brigade whose adequate manning shall be justified. The brigade shall consist of at least one full-time fire foreman and three full-time fire fighters (1 + 3). The plant fire brigade shall be at five minute (5) response preparedness at all times (7/24). The fire fighters shall be qualified in smoke diving in terms of training, experience, physical condition, suitability and equipment [30]. The plant fire brigade shall be equipped with a sufficient amount of suitable and efficient equipment.

The control room and the fire brigade shall be equipped with displays and printers for the fire detection system to speed up and facilitate identification of the fire locations and guidance to the scene of the incident.

Nuclear facilities shall be provided with equipment facilitating the use of a communication system generally in use by the authorities. Operation with the plant fire brigade and the regional fire and rescue services shall be planned, instructions provided and co-operation exercises conducted.

Overpressure Ventilation and Smoke Extraction:

The use of access routes between the control room and the emergency control room during fires shall be analysed and, where necessary, their reliability assured by special arrangements and taking into account the requirements of STUK's Guide YVL A.11.

Nuclear facilities shall be equipped with smoke extraction systems that remove the hot, possibly corrosive and toxic combustion gases generated by a fire:

- Rooms with heavy fire loads, such as the turbine hall and cable rooms, shall be provided with sufficiently efficient smoke extraction systems.
- The personnel carrying out fire extinguishing must be able to safely locate the fire.

2.5 Emergency Lighting

Emergency lighting shall be designed and installed at the nuclear facility comprising escape lighting as well as stand-by lighting for the control room, emergency control room, control centre and command centre. The emergency lighting shall enable safe passage inside the plant and escape from the buildings when normal lighting is out of order due to a disturbance in electricity supply, a fire or some other event.

2.6 Provision for Outages/Annual Maintenance

The nuclear power plant's design shall take into account plant servicing and maintenance. Fire protection shall make provision for fires occurring during outages by employing the defence in depth concept. Appropriate storage space, routes and instructions shall be in place for the storage and transport of temporary fire loads.

Additionally, STUK's Guide YVL B.7 provides guidelines for the plant layout design.

3. Fire Safety Assessment

3.1 General

Fire-induced failure shall be assessed by deterministic design methods in the first place and its significance for the nuclear power plant's safety shall be verified by a probabilistic fire risk assessment (Fire PRA) in accordance with STUK's Guide YVL A.7.

To verify the adequate implementation of the defence in depth principle in fire protection, fire hazard analyses shall be conducted including, e.g.:

- Fire simulations to evaluate fire propagation and the ambient effects of fire, temperature increase in particular.
- Analyses of heating, load-bearing capacity and integrity of load-bearing and separating structures.
- Analyses or calculations of temperature increase of components.

In risk-informed planning and assessment of fire protection, the results of deterministic fire hazard analyses shall be collected on a case-by-case basis and the adequacy of the nuclear facility's defence in depth ensured by accident modelling methods. The methods shall be used to assess the significance of fire protection impairments for fire safety at the nuclear facility.

Appendices to STUK's Guide YVL A.11 present procedures for providing protection against an aeroplane crash and STUK's Guide YVL B.7 presents requirements for layout design in accidents. In regards to the related fire consequences, the adequacy of fire protection shall be demonstrated by risk-informed design and fire hazard analyses.

3.2 Deterministic Fire Hazard Analyses

The adequacy of fire protection shall be demonstrated by deterministic fire hazard analyses. It is especially important to demonstrate that the safety functions of the facility can be reliably accomplished during any potential fire situation, thus all equipment in the fire compartment shall normally be assumed to fail due to a fire. In analysing the scope of consequential failures, the effects of smoke and other combustion gases shall be taken into account. The accomplishment of safety functions must be possible in accordance with the failure criteria of STUK's Guide YVL B.1. The reliable implementation of the safety functions of the facility shall not be endangered by any single failure or deviation in fire protection arrangements (e.g. an open fire door, or fire dampers fail to close).

Fire hazard analyses shall also examine design basis extension events (common cause failures in systems related to fire protection, e.g. the fire detection system is inoperative, the fire extinguishing system does not start or operation of the plant fire brigade is delayed). The results of deterministic fire hazards analyses are used as input data in drawing up a fire PRA.

Nuclear power plant design shall make provision for fire-induced initiating events and safety functions whose actuation is required during fire situations. Even if a fire at the nuclear power plant does not directly lead to an initiating event involving an automatic initiation of safety functions, provision shall always be made for promptly bringing the facility to a safe state during a fire situation in accordance with the operating procedures for transients and accidents.

It shall be demonstrated by a fire hazard analysis of the containment that the reactor can be shut down and cooled, and residual heat can be removed without compromising containment integrity.

It shall be demonstrated by a fire hazard analysis of the control room that control of the necessary safety functions can be executed in the event of a fire in the control room or in any other fire compartment. In connection with the design of the I&C systems of the nuclear power plant, the influence of fires on the functioning of safety significant I&C systems shall be analysed, including the effects of fire-induced temperature rise and combustion gases on equipment and the reflection of disturbances and failures thereof on the execution of safety functions.

Fire situations where a transformer fire or a switchgear fire potentially causes the simultaneous loss of all connections to the national grid shall be analysed and the results of the analysis taken into account in the design of grid connections. STUK's Guide YVL B.1 presents design requirements for the national grid connections of nuclear power plants.

The load-bearing capacity of the building frame (R) as well as the integrity (E) and insulation (I) of the separating structural elements shall be demonstrated in accordance with the fire resistance class requirements specified in the regulations and guidelines of the National Building Code of Finland [5], or by fire simulation and structural analysis.

3.3 Probabilistic Fire Risk Assessment (Fire PRA)

STUK's Guide YVL A.7 applies to fire risk analyses conducted during design, construction and operation of the nuclear power plant in order to assess the adequacy of fire protection and to identify fire-induced risk factors. The Fire PRA shall assess all fire events possibly inducing an initiating event.

The effects of malfunctioning fire water and fire extinguishing systems on the reliability of fire protection as well as the flood risk caused by the malfunctions shall be assessed in accordance with STUK's Guide YVL A.7. The adequacy of the testing methods of the fire detection and fire extinguishing systems shall be assessed by means of fire PRA.

STUK's Guide YVL A.7 prescribes that a design stage Level 1 and Level 2 PRA including a PRA computer model shall be drawn up for the review of the nuclear power plant's construction license application. The licensee shall supplement and update the Level 1 and Level 2 PRA including the PRA computer model, for the review of the nuclear power plant's operating license application.

4. Fire Safety During Operation

4.1 General

In the inspection and operation of nuclear facilities, the licensee shall take into account the fire safety requirements and aspects whose objective is to:

- Prevent outbreaks of fire.
- Rapidly detect and extinguish ignited fires.
- Prevent fire spread so that the facility's safety functions can be reliably performed also during a fire situation.

The licensee has overall responsibility for the development of the nuclear facility's fire safety and the maintenance of all fire protection arrangements. Fire safety requirements shall be taken into account in every field of operation. Everyone working at the facility is responsible for ensuring fire safety. For this purpose, there shall be training and instructions for both permanent and temporary facility personnel, and they shall be provided with adequate fire protection instructions.

4.2 Operational Limits and Conditions (OLC), Periodic Inspections and Maintenance

The licensee is responsible for maintaining fire protection arrangements in accordance with the procedures of a valid OLC and of the periodic inspection programme for fire protection. Any revisions by the licensee of the OLC's periodic testing programme are subject to approval by STUK.

If the original functional principles of fire protection systems, structures or components are changed or new systems or parts thereof are built, the plans for the changes shall be submitted to STUK for approval.

In disconnecting fire protection systems (fire detection and alarm systems, fire water systems or fire extinguishing systems) covered by the OLC for work carried out at the plant, the procedure shall be carried out in accordance with the OLC as well as approved plans and instructions. STUK's Guide YVL A.6 presents requirements related to OLC.

STUK shall be informed in advance of any significant/long-term repairs of fire protection systems. At the same time, any compensating measures to maintain the safety level prescribed in the OLC shall be presented.

In making essential changes to operative fire fighting preparedness, STUK's approval shall be obtained for the changes.

Fires and explosion events at the plant site, or situations involving a risk of them, shall be reported in accordance with STUK's Guide YVL A.10.

4.3 Nuclear Power Plant Outages

According to the OLC, components, structures and systems required for fire protection shall be operable also during nuclear power plant outages. The functionality and adequacy of fire protection arrangements shall be evaluated as part of outage planning. Outage specific special arrangements shall be undertaken to ensure adequate fire safety, where necessary.

A general description of refuelling outages and pre-planned extensive repair and maintenance outages shall be submitted to STUK for information no later than one month before the commencement of the outage. The description shall include arrangements to intensify fire protection arrangements during the outage.

Operative fire fighting preparedness during outages shall be intensified. During outages, a sufficient number of personnel with fire guard training shall supervise hot work and the fire protection arrangements.

The opening of separating penetrations and disconnecting of fire detection systems and extinguishing systems shall be done according to clearly defined procedures. Protective measures concerning hot work shall be determined in the work permit. Hot work can only be performed by those having a valid hot work permit in accordance with the work in question.

Hot work and other work presenting a fire hazard shall be provided with unambiguous instructions and supervision. For this purpose, training shall be provided to permanent and temporary power plant personnel and adequate instructions shall be available. If combustible liquids or gases are temporarily needed in rooms important to safety (e.g. for the purpose of decontamination or hot work), the amounts in question shall be the smallest possible and they shall be appropriately stored and kept, taking fire safety into account.

Before the nuclear power plant is restarted after an annual refuelling outage or a maintenance or repair outage of a longer duration, the licensee shall, as regards the fire protection arrangements, ensure that:

- Annual inspections required in the OLC have been performed.
- Structural fire protection meets the requirements of the OLC.
- Fire detection systems are operable.
- Fire extinguishing systems are operable.
- Access routes are open and housekeeping is on a high level at the facility.
- Temporary fire loads during the outage have been removed or stored safely in accordance with plans.
- Plant fire brigade is in normal preparedness and its equipment is in order.

4.4 Development of Fire Safety

The maintenance, assessment and continuous enhancement of fire safety shall be part of the safety culture relating to the operation of nuclear power plants. As part of the maintenance and development of fire safety, the fire PRA described in STUK's Guide YVL A.7 shall be kept up-to-date.

Fire hazard analyses and other documents shall be updated if conditions at the plant change or plant modifications are made to the plant's fire protection arrangements. New research results in the fire field, general progress in the field, accumulated knowledge of fire events as well as the ageing effects of components and materials shall be taken into account in fire hazard analyses. The before mentioned matters shall also be taken into account in plant operation and inspections as well as in personnel training.

5. Regulatory Oversight by the Radiation and Nuclear Safety Authority

STUK's inspections of fire protection at nuclear power plants and nuclear facilities are timed in accordance with the stages of the licensing process:

- During the decision-in-principle stage, STUK’s statement on an application for a decision-in-principle also covers the principles of fire protection.
- During the construction stage, STUK evaluates the Preliminary Safety Analysis Report (PSAR) and the supplementary topical reports, system descriptions, fire compartmentation drawings as well as the preliminary design and quality assurance procedures. The acceptability and feasibility of implementation of the fire protection principles are verified based on them. During the construction license phase, STUK also reviews the plant’s design phase fire PRA.
- During construction STUK ensures that the principles presented in the construction license stage are carried out in the plant’s detailed design and implementation. STUK oversees and inspects the plant construction in accordance with the construction inspection programme.
- During the operating license stage, STUK inspects the Final Safety Analysis Report (FSAR) and related system descriptions, the fire PRA and topical reports including the final analysis reports and commissioning inspection records drawn up by the license applicant and inspection bodies approved by Tukes.
- STUK conducts the commissioning inspections of fire protection systems as part of the commissioning inspections of buildings before the plant’s commissioning.
- The above stages may also apply to significant design modifications.
- The license for dismantling fire protection arrangements in relation to the decommissioning of a nuclear facility is provided by a separate decision.

STUK applies, as necessary, nuclear and radiation safety related fire research. For the purposes of document review, STUK may conduct or commission research work and expert assessments, such as:

- Assessment of the applicability of the whole comprising the design criteria and the applicable regulations and guidelines.
- Comparative risk and fire hazard analyses.
- Fire experiments.

STUK oversees and inspects the facility’s fire protection, condition monitoring and maintenance in conjunction with the inspections included in its in-service inspection programme and other inspections. At the same time STUK reviews the results of periodic inspections conducted by the licensee and other organisations. Furthermore, STUK oversees on-site as it deems necessary the periodic inspections conducted by the licensee.

In the inspection of modification, maintenance and repair work plans, as well as the actual construction, STUK follows the same process as with the approval of the original work, where applicable.

Where necessary in the handling of matters related to fire protection, STUK co-operates with other authorities, including the regional rescue services and the municipal building inspection authority.

Fire protection is also addressed in connection with emergency preparedness matters, where necessary.

STUK exchanges experiences with nuclear facility insurers who comply with Section 23 of the Nuclear Liability Act (493/2005) and arranges joint inspections, where necessary. Organisations that provide insurance cover for nuclear facilities issue international guidelines on fire protection at nuclear power plants (e.g. [31]).

6. Next Step: Rules for Application

The publication of STUK's Guides does not, as such, alter any previous decisions made by STUK. Hearing of the parties concerned is ongoing and STUK will issue a separate decision as to how the revised Guide B.8 applies to operating nuclear facilities or those under construction, and to licensees' operational activities. However, all new/revised Guides shall apply as they stand to all new nuclear facilities.

When considering how the new safety requirements laid down in the YVL Guides apply to operating nuclear facilities, or to those under construction, STUK takes into account Section 7 a of the Nuclear Energy Act (990/1987):

- The safety of nuclear energy use shall be maintained at as high a level as practically possible.
- For the further development of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology.

According to Section 7 r (3) of the Nuclear Energy Act:

- The safety requirements of STUK are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations.
- If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety standards in accordance with this Act, STUK may approve procedure or solution by which the safety level set forth is achieved.

References

[1] Nuclear Energy Act (990/1987)

[2] Nuclear Energy Decree (161/1988)

[3] Government Decree on the Safety of Nuclear Power Plants (717/2013)

[4] Government Decree on the Safety of Disposal of Nuclear Waste (736/2008)

[5] Ministry of the Environment, The National Building Code of Finland (RakMK)

[6] Rescue Act (379/2011)

[7] Government Decree on Rescue Services (407/2011)

[8] Rescue Equipment Act (10/2007)

[9] Government Decree on Equipment and Protection Systems Intended for Use in Potentially Explosive Atmospheres (917/1996)

[10] Decision of the Ministry of Trade and Industry on Equipment and Protective Systems Intended for Use in Potentially Explosive Atmospheres (918/1996)

[11] Government Decree on the Prevention of Danger for Workers Caused by Explosive Atmospheres (576/2003)

[12] Finnish Safety and Chemicals Agency (Tukes), Ministry of Social Affairs and Health, Department for Occupational Safety and Health, ATEX Safety of Explosive Spaces, 2003

[13] Land Use and Building Act (132/1999)

[14] Land Use and Building Decree (895/1999)

[15] IAEA SSR-2/1, Safety of Nuclear Power Plants: Design, 2012

[16] IAEA SSR-2/2, Safety of Nuclear Power Plants: Commissioning and Operation, 2011

[17] IAEA GS-R-2, Preparedness and Response for a Nuclear or Radiological Emergency, 2002

[18] IAEA GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, 2011

[19] IAEA GSR Part 4, Safety Assessment for Facilities and Activities, 2009

[20] IAEA NS-G-1.7, Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants, 2004

[21] IAEA NS-G-2.1, Fire Safety in the Operation of Nuclear Power Plants, 2000

[22] IAEA Safety Report Series No. 10, Treatment of Internal Fires in Probabilistic Safety Assessment for Nuclear Power Plants, 1998

[23] Western European Nuclear Regulators' Association (WENRA), Harmonization of Reactor Safety in WENRA Countries, Issue S, Protection Against Internal Fires

[24] Ministry of the Interior Decree on Automatic Fire Extinguishing Equipment (SM-1999-967/Tu-33)

[25] SFS-EN 12259, Fixed Fire Fighting Systems, Components for Sprinkler and Water Spray Systems

[26] CEA 4001, Sprinkler Systems, Planning and Installation

[27] CEA 4007, CO2 Systems, Planning and Installation

[28] CEA 4008, Fire Extinguishing Systems Using Non-Liquified Inert Gases, Planning and Installation

[29] CEA 4045, Fire Extinguishing Systems Using Liquified Halocarbon Gases, Planning and Installation

[30] Ministry of the Interior Directive for Rescue Diving (48/2007; SM050:00/2006)

[31] Nuclear Pool, International Guidelines for the Fire Protection of Nuclear Power Plants, 2006

Regulatory Framework and Insights from Fire Probabilistic Safety Assessment of Canadian Nuclear Power Plants

Usha Menon[1] and Yolande Akl[2]
Canadian Nuclear Safety Commission, Ottawa, Canada

Abstract

Recognizing that Probabilistic Safety Assessment (PSA) is a useful tool for ensuring the safety of Nuclear Power Plants, the Canadian Nuclear Safety Commission (CNSC) has incorporated PSA into the regulatory framework. The regulatory document S-294 "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" was issued by the CNSC in 2005. This regulatory document was amended to add recommendations from CNSC's Fukushima Task Force Report. The amended regulatory document has been approved and is in the process of being published as REGDOC 2.4.2 -"Probabilistic Safety Assessment (PSA) for Nuclear Power Plants".

As per S-294, Canadian licensees are required to perform Level 1 and Level 2 PSA for both internal and external events for at-power and shutdown plant states. The PSA results are used to identify weaknesses in the overall plant design for defence-in-depth, and to evaluate the benefits of plant modifications for improving operations and safety. PSAs are also used as an important input for accident management. In addition to other techniques, PSA is used as an input to an Integrated Safety Review in support of plant life extension and plant refurbishment projects.

For example, Fire PSA using probabilistic models analyzes internal fire events at various plant locations and evaluates the safety of the Nuclear Power Plant (NPP). It accounts for the various detection and suppression methods, and assesses the effects on, and damage to the equipment and cables that are required to maintain the plant in a safe state. It can provide the contribution of fire events to the overall plant risk.

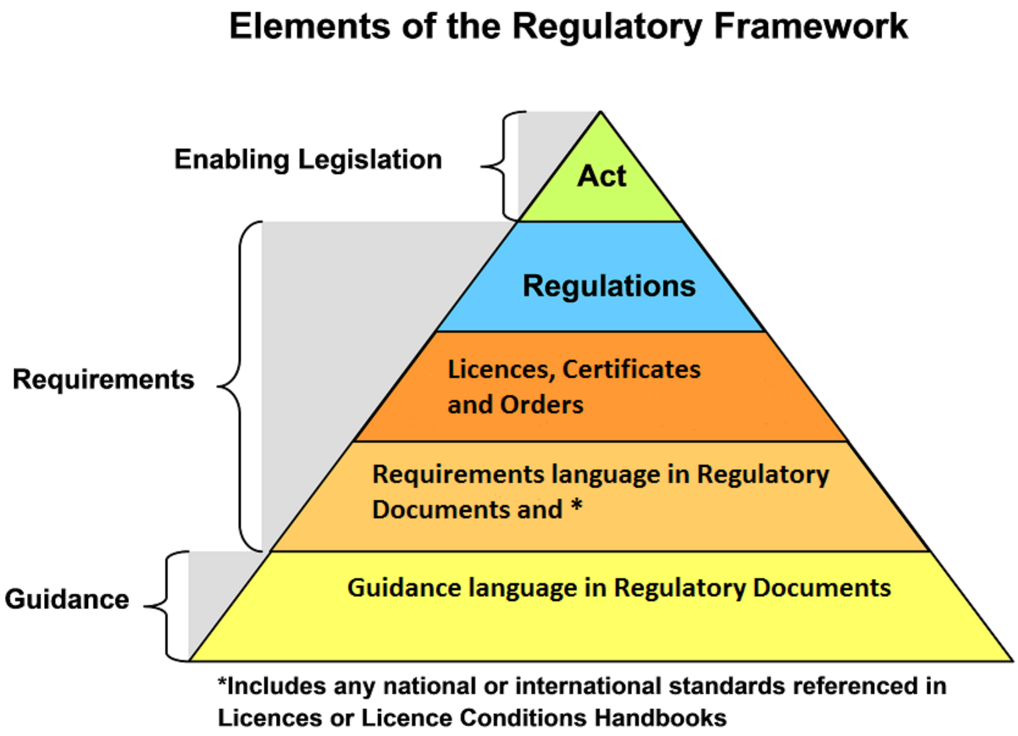
This paper presents some of the results and insights gained from the regulatory review of a Fire PSA conducted at a Canadian NPP as part of a refurbishment project. It also touches upon the current methodologies by Canadian licensees to conduct Fire PSA as part of compliance with the regulatory standard S-294 and re-licensing requirements.

1. Regulatory Framework in Canada

The Canadian Nuclear Safety Commission regulates the use of nuclear energy and materials to protect health, safety, security and the environment, and to implement Canada's international commitments on the peaceful use of nuclear energy.

1. *Technical Specialist, Probabilistic Safety Assessment and Reliability Division, CNSC, 280 Slater Street, Ottawa, Ontario, K1P 5S9, Canada, usha.menon@cnsc-ccsn.gc.ca*
2. *Director, Probabilistic Safety Assessment and Reliability Division, CNSC, 280 Slater Street, Ottawa,, Ontario, K1P 5S9, Canada, yolande.akl@cnsc-ccsn.gc.ca*

The diagram below shows the elements of the regulatory framework such as the enabling legislation, requirements and guidance.



The CNSC's regulatory framework usually present both requirements and guidance in a single document and distinguish between both through the use of mandatory (e.g., shall, must) and non-mandatory (e.g., should, may) language.

CNSC is committed to developing requirements and guidance that are technology-neutral to the extent practicable while adopting or adapting knowledge and best international practices such as those of the International Atomic Energy Agency (IAEA). In addition, CNSC’s work reflects the need to apply requirements and guidance to real-world scenarios commensurate with the risks presented by the licensed activities.

The CNSC has recently reorganized its regulatory documents in order to develop a sustainable structure that better reflects the CNSC's current approach to regulating the nuclear industry. The new structure re-organizes all existing documents and new document projects in a clear and logical manner, according to regulated facilities and activities, safety and control areas, and other areas of regulatory categories.

Domestic and international standards, in particular consensus standards produced by the Canadian Standards Association (CSA) Group, are an important component of the CNSC regulatory framework. Standards support the regulatory requirements established through the Nuclear Safety and Control Act (NSCA), its regulations and licences by setting out the necessary elements for acceptable design and performance at a regulated facility or a regulated activity. Standards are one of the tools used by the CNSC to establish the minimum acceptable levels of safety for design and operation of a nuclear facility. Licensees programs and practices are evaluated against the requirement of Codes and Standards referenced in the facilities licence conditions to determine if the licensees are qualified to carry out the licensed activity and will make adequate provisions to protect the health and safety of persons and the environment.

Recognizing that the Probabilistic Safety Assessment (PSA) is a useful tool for ensuring the safety of Nuclear Power Plants, the Canadian Nuclear Safety Commission (CNSC) has incorporated PSA in the regulatory framework. The regulatory document S-294 "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" was issued by the CNSC in 2005 [1]. This regulatory document was amended to add recommendations from CNSC's Fukushima Task Force Report. The amended regulatory document will be published as REGDOC 2.4.2 -"Probabilistic Safety Assessment (PSA) for Nuclear Power Plants".

As per S-294, the Canadian licensees are required to perform both Level 1 and Level 2 PSA for internal and external event for at-power and shutdown states. S-294 also requires Licensees to seek acceptance from CNSC for the methodology and Computer codes used for conducting the PSA.

S-294 has a footnote allowing the licensees, with the agreement of "persons authorized" by the Commission, to choose an alternative analysis method to conduct the assessment for external events.

2. Fire PSA in Canada

In Canada, PSAs are performed and submitted to CNSC for regulatory review to show compliance with S-294. The PSA results are also used to identify weaknesses in the overall plant design for defence-in-depth, and to evaluate the benefits of plant modifications for improving operations and safety. PSA also evaluates the operating procedures, and it is an important input for accident management. In addition to other techniques, PSA is used as input to an Integrated Safety Review in support of plant life extension and plant refurbishment projects.

Fire PSA using probabilistic models analyzes the internal fire events at various plant locations and evaluates the safety of the Nuclear Power Plant (NPP). It accounts for the various detection and suppression methods, and assesses the effects on, and damage to the equipment and cables that are required to maintain the plant in a safe state. It can provide the contribution of fire events to the overall plant risk.

3. Point Lepreau Refurbishment (PLR) Fire PSA

The Point Lepreau CANDU[3] (CANada Deuterium Uranium) pressurized heavy water (PHWR) reactor is owned and operated by New Brunswick Power (NB Power) and is located on Canada's east coast. It is a 700 MegaWatt electrical (MWe) class CANDU 6 reactor with a gross output of 680 MWe, supplying approximately 30 per cent of the province's electricity and first started operation in 1982.

The CANDU (CANada Deuterium Uranium) reactor is a Pressurized Heavy Water Reactor (PHWR) designed and built by Atomic Energy Canada Ltd. since the 1950s. The CANDU reactor uses heavy water as a moderator. The fuel bundles are placed in horizontal tubes (called pressure tubes). These tubes can be loaded remotely from either end while the reactor is running (on-line).

In 2008, NB Power began an extended outage to retube and refurbish the Point Lepreau reactor. During this process all 380 fuel channels and calandria tubes, along with the 760 feeder pipes were

[3] *CANDU (CANada Deuterium Uranium) is a registered trademark of Candu Energy Inc.*

replaced among other maintenance work. The station returned to commercial operation in November 2012 to deliver safe and reliable power to New Brunswick for the next 25 to 30 years.

The Level 1 and 2 Point Lepreau Refurbishment (PLR) fire PSA for at-power (also referred as full power) events was performed by Atomic Energy of Canada Limited (AECL now known as Candu Energy Inc.) along with support from the New Brunswick PSA team. The CNSC reviewed and accepted the Fire PSA methodology and performed a regulatory review of the PLR Fire PSA. Several periodic meetings and workshops were conducted with CNSC during the entire PLR project discussing the methodology to be applied as well as comments/feedback on the PSA reports before the CNSC acceptance.

3.1 Methodology, results and insights from Level 1 and Level 2 PLR Fire PSA

The main objective of conducting the PSA was to provide insights into plant safety design and performance, including the identification of dominant risk contributors and the comparison of options for reducing risk. The scope of the assessment covered Level 1 and 2 PSA and included internal events for full power and shutdown, and internal fires, floods, and seismic events for full power operation. This paper only covers the results and insights from the L1 and L2 PLR internal fires full power PSA [2, 11] which was submitted to CNSC as part of the licensing submission for plant refurbishment activity to show compliance with S-294.

3.1.1 Methodology for PLR Fire PSA

The Level 1 PSA for internal fires estimated the fire-induced severe core damage frequency (SCDF) and the Level 2 estimated the large release frequency (LRF). The Level 1 fire PSA analysis was conducted in two stages: fire analysis and accident sequence quantification (ASQ). In the first stage, fire analysis, fire characteristics are identified, namely fire ignition sources, combustibles, fire protection barriers, and location of PSA credited components and cables. Subsequently, fire scenarios are established based on the physical proximity of fire barriers, ignition sources, combustibles and plant procedures in the event of a fire. A plant walk down is conducted for collecting this information and preparation of the fire database.

Qualitative screening is performed to eliminate fire scenarios from further analysis based on the location of safety related systems and equipment. For the fire scenarios that are not screened out qualitatively, the fire compartment ignition frequency is estimated using the fire ignition frequency database. Quantitative screening is performed to eliminate fire scenarios from further analysis by removing explicit conservatisms in the fire scenario prior to evaluating the SCDF. Fire scenarios that are not screened out are assessed quantitatively in the second stage ASQ.

For each retained fire scenario, the fire analysis provides the ASQ with a list of components that are damaged due to the fire. In the ASQ mitigating and support systems that are impaired by fire are identified and an event tree (ET) representing the plant response following the fire is developed. The ETs are quantified with a fault tree (FT) that accounts for components affected by fire (including cables through cable routing analysis) in order to estimate the conditional core damage probability (CCDP). The product of the CCDP and the fire scenario frequency yields the SCDF for the fire scenario.

For both Level 1 and Level 2 ASQ, recovery actions were applied at the sequence level. These actions may include the recovery of previously unavailable equipment, or the use of non-standard procedures to mitigate accident conditions.

The Unified Partial Method (UPM) was used to calculate Common Cause Failures (CCFs) [4]. UPM is a development of the Partial Beta-Factor method for assessing CCFs at a component level. UPM assumes that the beta-factor is influenced by eight underlying factors (Redundancy and

Diversity, Separation, Understanding, Analysis, Man Machine Interaction, Safety Culture, Environmental Control, and Environmental Testing). Each underlying factor is associated with a weight and a score. Dominant CCFs were re-evaluated using the alpha factor method and the USNRC database [5].

Human Reliability Analysis (HRA) was performed using Accident Sequence Evaluation Program methodology (ASEP) for the Level 1 PSA [6] and a simplified ASEP methodology for the Level 2 PSA [7], as well as accounting for operator dependency. Operator actions were evaluated, taking into account the existing abnormal plant operating procedures (APOP) and emergency operating procedures (EOP) for Point Lepreau as well as the operating manuals for the loss of other systems that are not covered by APOP/EOP, such as loss of moderator system and loss of end shield cooling. For Level 2 PSA, the operator actions were evaluated taking into account the existing Severe Accident Management Guidelines (SAMG). The dependency between operator actions in the same accident sequence was evaluated for Level 1 through recovery after Accident Sequence Quantification using the Standardized Plant Analysis Risk HRA (SPAR-H) method [8] and for Level 2 directly in the containment event trees (CETs). The dominant HRA contributors were re-evaluated using the Technique for Human Error Rate Prediction (THERP) methodology [9].

3.1.2 Results and insights from L1 and L2 PLR Internal Fire PSA

The analysis considered potential fires in three buildings separately: reactor building (R/B), turbine building (T/B), and service building (S/B). The T/B analysis includes the auxiliary service building while the S/B analysis includes the Secondary Control Area (SCA), Emergency Core Cooling (ECC) Building, Fresh Water Pumphouse (FWPH), and Condenser Circulating Water (CCW) Building.

Severe core damage (SCD) accidents are beyond-design-basis accidents in which a rapid or late loss of the structural integrity of the reactor core occurs. A loss of core structural integrity results from a loss of heat sinks leading to core damage involving multiple fuel channels failures and core disassembly. To estimate the severe core damage frequency (SCDF), the accident sequences leading to SCD were summed together. The fire-induced SCDF estimated from this ASQ is summarized in Table 1.

Originally, the Reactor building SCDF was dominated by the fire scenario in which an oil fire occurs in the dykes below the Primary Heat Transport pumps and spreads to moderator room via the opening of the dykes. A sensitivity case was performed to consider the effect of blocking the dyke openings to prevent the fire from propagating to the moderator room. The results demonstrated that the SCDF from this fire scenario was reduced from 1.55E-05 to 1.06E-07 events/year and therefore the R/B SCDF was reduced from 1.81E-05 to 2.66E-06 events/year. Based on the reduction to plant risk, the modification to the dykes was recommended. The T/B and S/B SCDF results were judged to be acceptable and therefore no recommendations were made. The fire-induced SCDF represents 32% of the target for SCDF for internal and external events.

The Accident Sequence Quantification was performed for all sequences for both Level 1 and Level 2 using a cut off truncation limit that is three orders of magnitude lower than the sequence frequency. From the Level 1 PSA results, 322 sequences accounting for 98% of the total severe core damage for full power were analysed in the Level 2 report and were grouped according to three representative severe core damage accidents: station blackout (SBO), small loss of coolant accident (SMLOCA) and in-core LOCA (INCRLOCA).

The results of the Level 1 analysis identified the representative scenarios for which severe accident progression analysis was performed using Modular Accident Analysis Program MAAP4-CANDU [10]. MAAP4-CANDU (M4C) is a computer code that can simulate the response of the CANDU6 plant during severe accident conditions, including actions that are undertaken as part of accident management.

Similar to internal events, the sequences within each of the three primary groups were further grouped according to similar plant configuration accounting for the status of mitigating systems, the impact on the containment system availability, and the containment status. Station Blackout sequences were grouped into 2 cases; Small LOCA sequences were grouped into 24 cases; and In-core LOCA sequences were grouped into 75 cases; yielding a total of 101 containment event trees [11].

The results as shown in Table 2 for SCDF and LRF for Point Lepreau NPP from internal fire events at full power was 2.59E-05 and 5.05E-07 events/year [11] which is lower than the target for SCDF 1E-04 events /year [3] and 1E-05 events/year [12] for LRF, which is in line with the international results.

Fires in the High Voltage Switchgear room, the Fuelling Machine Auxiliary Cleanup room, and the Main Control Room are the highest contributors to SCDF at full power [11].

The sequences with the highest contribution to the LRF at full power were due to fires. Fires in the Chemistry Lab are the highest contributor (due to hydrogen cylinders being stored under a main cableplenum) , followed by fires in the High Voltage Switchgear Room, the Reactor Building Ventilation Exhaust and Sampling Cabinets Room, the D_2O Vapour Recovery Room, and the Cable Access and Corridor Room. These five most dominant fires account for 69.3% of the LRF at full power. All fire events account for 83.7% of the LRF at full power [11].

Fire events dominate the results for both full power SCDF and LRF, representing 64.3% of SCDF and 83.7% of LRF [11].

The design changes at Point Lepreau NPP as a result of the fire PSA are given below:

- Upgrades to fire detection, such as very early smoke detection apparatus (VESDA) in R/B and S/B, new smoke early detection in R/B, S/B and T/B, new sprinklers in S/B and T/B.
- Installation of dyke to retain oil at PHT pump troughs.

4. Current Methodologies for Fire PSA in Canadian NPPs

According to S-294 the PSA models should reflect the plant as as-built and operated and updated every 3 years or sooner if major changes occur in the facility (now being proposed to be submitted every 5 years in the forthcoming REGDOC 2.4.2). Licensees have submitted the updated PSAs which includes the submission of Fire PSA as part of their compliance to S-294 for the re-licensing or plant life extension programme. The methodologies for Fire PSA are reviewed and accepted by CNSC. Some Licensees have requested an exemption to perform Shutdown state PSA for internal fires, flood and seismic events. CNSC has recently proposed the following approach based on the best international practices [15]:

1. Ensure defence in depth for outage such that the key safety functions are maintained
2. Ensure front line system availability for each Plant Operating State (POS)
3. Perform a systematic analysis based on the identified POS.

This approach comprises two parts for each external event:

I. Qualitative Risk Analysis (QLRA)

- Perform qualitative risk analysis using Defense-In-Depth (DID) principles to support risk assessment and management for shutdown operation.

II. Limited Quantitative Analysis

- Quantify the risk using available information through bounding or conservative assessments.

If the limited quantitative analysis finds the shutdown risk for a specific external event is too high to ignore, (i.e. if it cannot be shown using a demonstrably conservative analysis that the CDF is <1E-6/yr) a detailed analysis should be performed.

CNSC proposed approach does not exclude other approaches the licensees may propose. If other approaches are proposed, the licensee needs to demonstrate that their approaches have the same level of details compared with CNSC staff proposed approach. CNSC intends to include this approach in the future regulatory guidance document.

In general, the Licensees follow the ASME/ANS RA-S-2009 [21] and NUREG/CR- 6850 [22] guidelines to conduct the L1 internal Fire events full power PSA. Some licensees proposed a phased to conduct the Fire PSA. The phased approach develops the overall PSA in manner that focuses on the most risk significant areas and uses more conservative or screening approaches for non-risk significant contributors. The fire safe shutdown analysis and fire hazard analysis is used an important input to conduct these Fire PSAs.

CNSC is in the process of reviewing and accepting the methodologies and Fire PSAs submitted as part of the licensing of new builds, re-licensing and plant life extension activities for Canadian NPPs.

5. Conclusions

Regulatory Framework in Canada emphasises the role of PSA in risk informed decision making. CNSC has published the Regulatory Standard S-294 which requires the Licensees to submit Level 1 and Level 2 PSA for internal and external events during at-power and shutdown states and also seek acceptance of the methodology and computer codes used in PSA. This regulatory document was amended to add recommendations from CNSC’s Fukushima Task Force Report. The amended regulatory document has been approved and is in the process of being published as REGDOC 2.4.2 -“Probabilistic Safety Assessment (PSA) for Nuclear Power Plants”.

Results and insights from the Point Lepreau Refurbishment (PLR) Fire PSA identified Fire events as a dominant contributor to SCDF and LRF. Several design changes were implemented at Point Lepreau NPP based on the Fire PSA results to improve the safety of the NPP. For re-licensing activities in Canada the Licensees follow the international best practice for e.g.: NUREG/CR 6850 guidelines for conducting Fire PSA to show compliance to S-294. CNSC continues to review and accept the methodologies as well as the Fire PSAs as part of the licensing, re-licensing, plant life extension and plant refurbishment activities.

Acknowledgement

The authors would like to acknowledge Marcel de Vos, CNSC and Liliana Comanescu, Candu Energy Inc. for their valuable support in providing information for this paper.

Table 1. **Summary of Fire-Induced Severe Core Damage Frequency for PLR [2]**

Location	**Fire-Induced Severe Core Damage Frequency (events/year)**
Reactor Building	2.66E-06*
Turbine Building 1	1.81E-05
Service Building 2	1.16E-05

* - with proposed design changes

Table 2. **SCDF and LRF for PLR Fire PSA [11]**

Results	**Frequency Events/Year**
Fire-Induced Severe Core Damage	2.59E-05
Fire-Induced Large Release	5.05E-07

References

[1] CNSC, Regulatory Standard S-294, -“Probabilistic Safety Assessment (PSA) for Nuclear Power Plants”, April 2005

[2] Wei, M. et al., Level 1 External Events (Internal Fire, Internal Flood and Seismic Events) PSA Accident Sequence Quantification for Point Lepreau Refurbishment Project, PSAM9, Hong Kong, 2009

[3] IAEA, Basic Safety Principles for Nuclear Power Plants, Safety Series Document No. 75-INSAG- 3 Rev. 1, INSAG-12, 1999

[4] UPM 3.1: A Pragmatic Approach to Dependent Failures Assessment for Standard Systems, SRDA-R13, SRD Association, AEA Technology PLC, Cheshire, UK, 1996

[5] U.S. NRC, CCF Parameter Estimations, 2003 Update, http://nrcoe.inl.gov/results/CCF/ParamEst2003/ccfparamest.htm, May 2006 (accessed May 2007)

[6] U.S. NRC, Accident Sequence Evaluation Program: Human Reliability Analysis Procedure, NUREG/CR-4772, February 1987

[7] M. Richner, "Modeling of SAMG Operator Actions in Level 2 PSA", PSAM-0164, Proceedings of the 8th International Conference on Probabilistic Safety Assessment and Management, New Orleans, Louisiana, USA, May 2006

[8] U.S. NRC, The SPAR-H Human Reliability Analysis Method, NUREG/CR-6883, August 2005

[9] U.S. NRC, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, August 1983

[10] MAAP4-CANDU Modular Accident Analysis Program for CANDU Power Plant, Version 4.0.5+, Volume 1: User Guidance, Fauske & Associates Inc., April 1998

[11] Comanescu, L., et al., Point Lepreau Refurbishment Project Level 2 PSA results for internal and external events, Paper presented at ANS PSA 2008 Topical Meeting - Challenges to PSA during the nuclear renaissance, Knoxville, Tennessee, September 7–11, 2008

[12] IAEA, Development and application of Level1 Probabilistic Safety Assessment for Nuclear Power Plants, SSG-3, 2010

[13] IAEA, Determine the quality of Probabilistic safety Assessment (PSA) for applications in Nuclear Power Plant, TECDOC-1151, 2006

[14] IAEA, Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes, TECDOC-1144, 2000

[15] EPRI, Qualitative Risk Assessment Methods for Shutdown Risk Management, EPRI-1013501, November 2006

[16] USNRC, Draft Methodology for Low Power/Shutdown Fire PRA, NUREG/CR-7114, December 2011

[17] OEAD/NEA, Improving Low Power and Shutdown PSA Methods and Data to Permit Better Risk Comparison and Trade-off Decision Making, NEA/CSNI/R(2005)11/VOL1, September 2005

[18] OEAD/NEA, Low Power and Shutdown Operations Risk: Development of Structure for Information Base and Assessment of Modelling Issues, NEA/CSNI/R (2009)17, December 2009

[19] USNRC, Evaluation of Potential Severe Accidents During Low Power and Operations at Grand Gulf, Unit 1, NUREG/CR-6143, 1995

[20] USNRC, Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1, NUREG/CR-6144, 1995

[21] USNRC, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Faciliies, NUREG/CR-6850/EPRI 1011989, Volumes 1 and 2, September 2005

[22] ASME, ANS, Addenda to ASME/ANS RA-S-2008 Standard for Level1/ Large Early Release Frequency PRA for Nuclear Power Plants, ASME/ANS RA-S-2009, New York, February 2009

Guidance for Performing Fire PRA in Germany

Heinz Peter Berg[a],
[a] Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany

Abstract

Fire PRA is required as part of the probabilistic safety assessment within periodic safety reviews (PSR) in Germany. Fire PRAs have been conducted for all nuclear power plants in Germany within the second PSR. Thus, Fire PRA has become an additional tool to supplement deterministic assessment of the fire protection for supporting decision making.

However, according to the German PSA Guideline and its corresponding technical documents on PSA methods and data issued 2005, Fire PRA has been focused on full power plant operational states. In the frame of the second PSR the scope was extended to low power and shutdown states, bur not assessing internal and external hazards.

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

1. The total compartment inventory of systems, structures and components (SSC) including cables has to be associated to each plant operational state (POS).
2. Depending on the POS, changes of the status with regard to fire protection in certain compartments are possible.
3. In case of a fire induced loss of the entire compartment inventory, the list of potential initiating events should be adapted and, if necessary, further completed depending on the respective POS.

During LPSD so-called transient fire loads and/or additional ignition sources are temporarily present in those rooms where these are needed for maintenance and repair activities including hot work. An evaluation of the hot work permits is particularly needed for observing and considering potential peculiarities during these activities typically performed during LPSD from the beginning of the analysis and being able to consider those time periods explicitly for the fire occurrence frequency estimation.

The extension of Fire PRA to LPSD states has to be particularly applied to those plants, for which a third PSR is required by the German regulations. In addition, the safety significance of modifications in the plant important to safety should be evaluated for which a significant influence on the PRA results can be expected. This also covers to demonstrate that the necessary fire safety measures are sufficient.

1. Introduction

Operating nuclear power plants (NPPs) in Germany have been designed and constructed in different plant generations resulting in differences in the design and layout of fire protection features. Thus, it was necessary to assess the current fire safety status of the nuclear power plant and if this status is sufficient.

In the past the safety concept of NPPs as well as licensing decisions by the competent authorities and their experts in Germany were mainly based on deterministic principles such as prevention or control of abnormal plant operational conditions and incidents by technical means to ensure high reliability. In the meantime, probabilistic safety analyses (PSA) are mainly applied in the frame of periodic safety reviews and shall supplement deterministic safety demonstrations to verify the balance of the plant design related to safety.

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

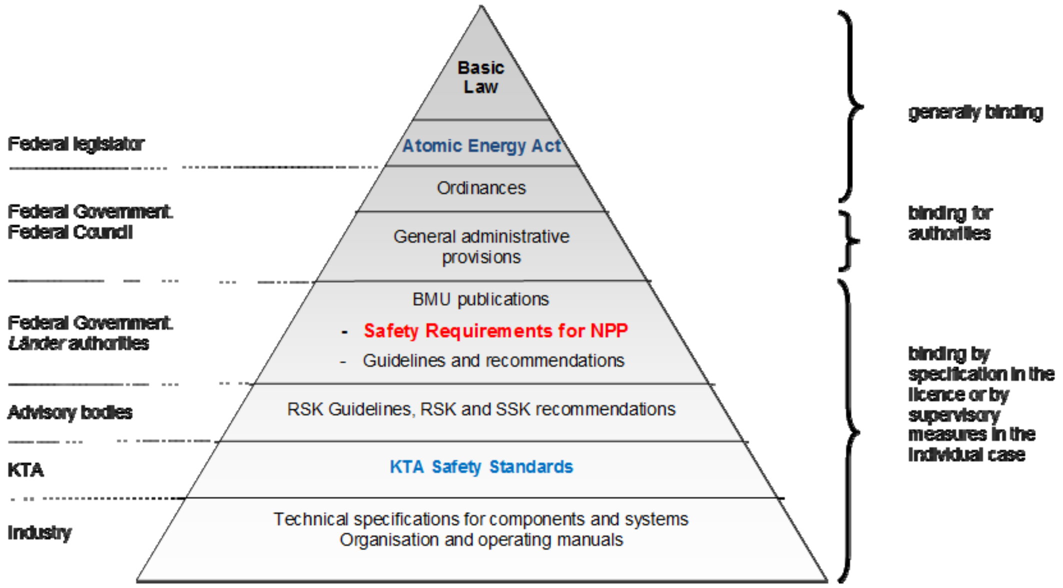


Figure 1. **Nuclear regulatory framework in Germany**

The German nuclear regulatory framework has recently been significantly enhanced promulgating state-of-the-art "Safety Requirements for Nuclear Power Plants" issued by the Federal Ministry of the Environment, Nature Conservation and Reactor Safety [1]. It is systematically structured and follows the safety approach of defense-in-depth coincident to the safety principles of IAEA SSR-2/1 [2].

These enhancements also address the adequate consideration of internal as well as external hazards, in particular fires, in particular with regard to event combinations of fires and other anticipated events.

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

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

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

- Plant internal flooding,
- Plant internal fires and explosions,
- Increased radiation levels,

…

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

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

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

for which a significant influence on the results of the PSA can be expected.

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

2. Deterministic Fire Safety Assessment

As already indicated, in Germany the NPP safety concept and the licensing decisions by the regulatory authorities and their experts in charge were based on deterministic safety principles, such as prevention and control of abnormal plant operational conditions and incidents by various technical in-depth means such as passive barriers, redundancy and diversity of safety systems to ensure high reliability.

Moreover, safety decision making within the design and licensing process was principally based on demonstrating compliance with pre-described technical requirements as provided, e.g., in the German nuclear safety standards issued by the Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). On the level of technically detailed KTA standards in total three standards are available with respect to fire safety:

- KTA 2101.1: Fire Protection in Nuclear Power Plants, Part 1: Basic Requirements [3],

- KTA 2101.2: Fire Protection in Nuclear Power Plants, Part 2: Fire Protection of Structural Plant Components [4],
- KTA 2101.3: (Fire Protection in Nuclear Power Plants, Part 3: Fire Protection of Mechanical and Electrical Components [5].

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

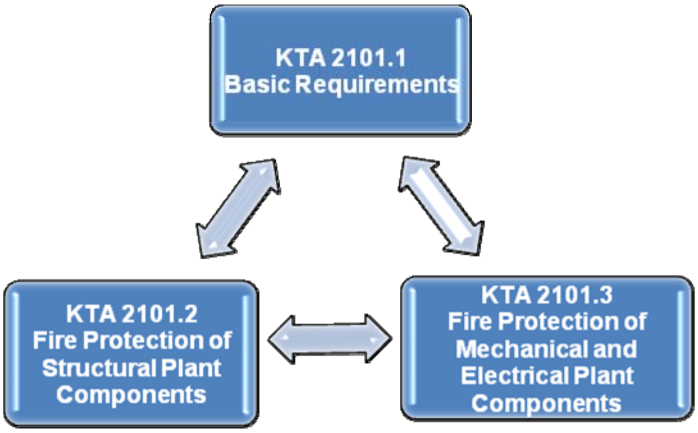


Figure 2. **German nuclear safety standards KTA 2101 “Fire Protection in Nuclear Power Plants”**

Major goals of the update of KTA 2101, Part 1-3 are the following:

- Adapting requirements to the actual state-of-the-art
  - corresponding to the most recent, also non-nuclear standards and norms,
  - providing specific compliance with requirements regarding the fire brigade,
  - considering low power and shutdown plant operational states better and more systematically, and
  - covering event combinations of fires and other anticipated events more systematically (post-Fukushima requirement).
- Compliance with the “Safety Requirements for NPP” [1] regarding the following aspects:
  - Better consideration of the defense-in-depth concept,
  - Specific compliance with requirements for the safety demonstration, in particular requiring
    - a systematic and comprehensively documented deterministic fire hazard analysis to be kept up to date,

  - an adequately documented probabilistic fire risk analysis (Fire PSA) for all plant operational states (full power as well as intended as well as unintended low power and shutdown states including the post-commercial shutdown phase of longer duration) to be kept up to date, and
  - deterministic safety demonstrations supplemented by probabilistic assessment in case of plant modifications.

- A more systematic approach and outline of the standards covering only nuclear specific requirements and deviations from non-nuclear standards and norms, in particular with respect to combinations of fires with other anticipated events have to be assumed, if they occur as consequence of the initial event or if their occurrence at the same time has to be accounted for due to their occurrence frequency and the extent of damage). In this context, the following event combinations have to be considered:
  - Fire and consequential event,
  - Anticipated event and consequential fire, and
  - Fire and independently occurring anticipated event.

The recent update of the KTA 2101, Part 1 to 3, is ongoing. The enhanced standards are intended to be issued in 2014.

3. Guidance for Fire PSA

The fundamental boundary conditions for performing PSA including requirements with respect to their documentation are provided in [6]. This PSA Guideline contains reference listings of initiating events for nuclear power plants with PWR (pressurized water reactor) and BWR (boiling water reactor) respectively, which have to be checked plant specifically with respect to applicability and completeness. Plant internal fires are included in these listings.

Detailed instructions for the analysis of plant internal fires, fire frequencies and unavailability of fire detection and alarm features as well as data, e.g., on the reliability of active and passive fire protection means are provided in the technical documents on PSA methods [7] and PSA data [8]. However, the Fire PSA guidance was limited to the full power operational state.

For the fire risk assessment, different screening approaches are applied to identify critical fire compartments. The models proposed have been successfully applied in several fire risk studies for German NPPs.

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

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

If a complete plant specific PSA is available, the fire induced hazard states frequencies will be summarized for all initiating events and specified as input to the corresponding event tree of the Level 1 PSA.

Furthermore, the plant hazard states have to be introduced into the fault trees. The plant hazard state frequencies are estimated for each transient as the sum of the single event core damage frequencies. The total plant hazard state frequency is obtained by summarizing the contributions of all transients. Moreover, the fire induced core damage frequency has to be calculated for the full power operational state.

The safety requirements for NPPs in [1] further require that state-of-the-art methods, models and data have to be applied in the frame of PSA. As far as possible plant specific data have to be applied. If no suitable plant specific data are available from the operating experience, generic data may be used if justified. Such data are provided for fire protection features in [9].

In the meantime, the guidance for performing fire PSA for low power and shutdown (LPSD) plant operational states (POS) has been drafted and is outlined below.

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

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

The guidance for LPSD plant operational states will be part of a supplementary document to [7] and [8], which is intended to be issued in 2014.

In the context of modelling plant specific fire event and fault trees, reliability data with regard to fire protections means are required. In [10] technical reliability data for various active fire protection features have been provided resulting from plant specific analyses of the operating experience from different NPPs.

In order to update the already existing reliability data (in particular, failure rates per hours of plant operation) as well as to extend the database covering an additional plant unit the following components and systems installed in six NPPs at five plant sites have been investigated:

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

The evaluation has been performed by analyzing the documentation of periodic in-service inspections as well as additional information and reports which resulted from the inspection findings. In case of more complex systems such as the fire detection systems, fault trees are presented to calculate the system's reliability in addition to the components' reliability data.

These data are then used in a second step to provide generic data, only based on German operational experience.

The updated and extended generic database covers 111 plant operational years of in total six German NPP units of different type (PWR as well as BWR) and age. These enhancements have increased the level of confidence significantly. More details regarding the derivation of the new data set are provided in [9] and [10].

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

Also the new set of reliability data for fire protections means will be part of the supplementary document mentioned above.

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

Compared with the unchanged condition of the plant, modifications of measures, equipment or the operating mode of the plant must not lead to an increase in the average core damage frequency (CDF) and the average frequency of large and early releases (LRF/LERF), neither for power operation nor for low-power and shutdown states, considering all plant-internal events as well as all internal and external hazards as well as very rare human induced external hazards.

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

The CDF values have been calculated in the frame of the comprehensive (periodic) safety reviews. The results of the latest safety review for the respective NPP are the basis for the comparison in case of modifications.

This requirement in [1] results in the extended use of PSA to regulatory issues beyond PSR, such as a regulatory oversight on modifications applied by the licensee or an evaluation of those safety relevant events

Including probabilistic considerations in the assessment is particularly useful in the evaluation of decision alternatives and strategies. A typical application in the past was the assessment of modifications with respect to in-service inspection intervals.

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

4. Concluding Remarks

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

Meanwhile, fire PSA is required in Germany as part of the PSA within periodic safety reviews. Therefore, fire PSAs have been conducted for all nuclear power plants in Germany within the second PSR and are seen as an additional tool to supplement deterministic assessment of the fire protection for supporting decision making.

The requirement to use only qualified PSA codes has also to be met for Fire PSA. Moreover, validated fire simulation models and codes have to be used in case of deterministic fire hazard analysis and probabilistic fire risk analysis and assessment.

It has to be stated that according to the German PSA guideline [6] and its corresponding technical documents on PSA methods [7] and data [8], Fire PSA has been focused on full power plant operational states in the past. In the frame of the second PSR. the scope was extended to low power and shutdown states for internal events but not fully extended to internal hazards. Fire PSA for LPSD has to be performed for those plants, for which a third safety review is required to be conducted due to the German regulations [12]. Guidance for LPSD plant operational states will be part of a supplementary document to [7] and [8] intended to be issued in 2014.

The recently issued state-of-the-art “Safety Requirements to Nuclear Power Plants” [1] also underline the need for an adequate fire protection design and the demonstration of the reliability of the selected equipment by deterministic and probabilistic safety assessments.

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

This was one of the reasons to continue the update of the three parts of the German nuclear safety standard KTA 2101, Part 1-3 with respect to fire protection. It is intended to issue these standards at the end of 2014.

Moreover, they concern safety demonstrations by deterministic as well as probabilistic safety assessment in more detail. In this context, PSA shall supplement deterministic safety demonstrations with regard to the balance of the safety related plant design. PSA shall also supplement deterministic safety demonstrations to assess the safety significance

- of modifications of measures, equipment or the operating mode of the plant, as well as

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

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

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

References

[1] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) (2013), Safety Requirements for Nuclear Power Plants, Federal Gazette, January 24, 2013 (in German), http://www.bmu.de/service/publikationen/downloads/details/artikel/bekanntmachung-der-sicherheitsanforderungen-an-kernkraftwerke-vom-22-november-2012/?tx_ttnews%5BbackPid%5D=266

[2] International Atomic Energy Agency (IAEA) (2012), Safety of Nuclear Power Plants: Design, Specific Safety Requirements, IAEA Safety Standards Series No. SSR-2/1, Vienna, Austria, January 2012, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1534_web.pdf

[3] Nuclear Safety Standards Commission (KTA, German for: Kerntechnischer Ausschuss) (2000), KTA 2101.1 (12/2000), Fire Protection in Nuclear Power Plants, Part 1: Basic Requirements (Brandschutz in Kernkraftwerken, Teil 1: Grundsätze des Brandschutzes), Safety Standards of the Nuclear Safety Standards Commission (KTA), December 2000, http://www.kta-gs.de/e/standards/2100/2101_1e.pdf

[4] Nuclear Safety Standards Commission (KTA, German for: Kerntechnischer Ausschuss) (2000), KTA 2101.2 (12/2000), "Fire Protection in Nuclear Power Plants, Part 2: Fire Protection of Structural Plant Components (Brandschutz in Kernkraftwerken, Teil 2: Brandschutz an baulichen Anlagen)", Safety Standards of the Nuclear Safety Standards Commission (KTA), December 2000, http://www.kta-gs.de/e/standards/2100/2101_2e.pdf

[5] Nuclear Safety Standards Commission (KTA, German for: Kerntechnischer Ausschuss) (2000), KTA 2101.3 (12/2000), Fire Protection in Nuclear Power Plants, Part 3: Fire Protection of Mechanical and Electrical Components (Brandschutz in Kernkraftwerken, Teil 5: Brandschutz an maschinen- und elektrotechnischen Anlagen), Safety Standards of the Nuclear Safety Standards Commission (KTA), December 2000, http://www.kta-gs.de/e/standards/2100/2101_3e.pdf

[6] Federal Ministry of the Environment, Nature Conservation and Reactor Safety (BMU) (2005), Guideline for Conducting the Safety Review According to § 19a of the Atomic Energy Act – Guideline on PSA - August 30, 2005, Federal Bulletin Nr. 207a, 2005

[7] Facharbeitskreis (FAK) PSA (2005), Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke. Stand: August 2005, BfS-SCHR-37/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, 2005 (in German only)

[8] Facharbeitskreis (FAK) PSA (2005), Daten für probabilistische Sicherheitsanalysen für Kernkraftwerke. Stand: August 2005, BfS-SCHR-38/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, 2005 (in German only)

[9] Forell, B., S. Einarsson, M. Röwekamp, and H.-P. Berg (2012), Updated Technical Reliability Data for Fire Protection Systems and Components at a German Nuclear Power Plant, in: 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012 (PSAM11/ESREL 2012), ISBN: 978-1-62276-436-5, Curran Associates, Inc., Red Hook, NY, USA, 2012, pp. 3783-3794

[10] Forell, B., S. Einarsson, and M. Röwekamp (2014), Technical Reliability of Active Fire Protection Features – Generic Database Derived from German Nuclear Power Plants, in: Proceedings 12th International Probabilistic Safety Assessment and Management Conference, Honolulu, HI, USA, in preparation, 2014

[11] Babst, S., M. Röwekamp, M. Schwarz, and M. Türschmann (2014), Conducting Fire PSA for the Post-commercial Shutdown Phase, in: Proceedings of the International Workshop on Fire PRA, Garching, Germany, 28 – 30 April 2014, to be published

[12] Atomic Energy Act (Act on the peaceful utilisation of nuclear energy and the protection against its hazard) (2013), December, 23, 1959, as amended and promulgated on July, 15, 1985, last amendment of August 28, 2013, Federal Bulletin I, p. 1565

Fire PSA Attributes in the Integrated IAEA Guidelines on PSA Quality

Artur Lyubarskiy, Irina Kuzmina and Peter Hughes
International Atomic Energy Agency, Vienna International Center, PO Box 100, 1400 Vienna, Austria
Dennis Henneke
GE Hitachi Nuclear Energy, 4200 Wildwood Parkway, Atlanta, Georgia, 30339, United States

Abstract

In 2006 the IAEA has published the TECDOC-1511 “Determining the Quality of Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants” [1]. It provides an approach and detailed guidance for achieving the technical quality of PSA needed to support various PSA applications, including the base case PSA. The document provides a comprehensive list of “General Attributes” that have to be satisfied to assure PSA quality for any PSA and “Special Attributes” required for enhanced quality, level of details and scope of PSA that are needed for particular PSA applications. The original TECDOC-1511 covers only internal events at full power. Since it has proven to be very useful for the Member States, it was determined that it should be expanded to cover all modes and all hazards. Therefore, IAEA initiated a project to expand the TECDOC. Currently the draft of the expanded TECDOC that covers internal fires and floods, external hazards, and low power/shutdown modes of operation is available. An important feature of the document is that the new technical information is integrated into the existing structure of the TECDOC-1511 instead of creating new sections for each hazard and mode of operation. The TECDOC puts the additional material unique to each hazard into the existing technical elements, therefore avoiding "back-references" and duplications of requirements in different parts. The Technical Meeting held on 6-10 December 2014 provided an opinion that the development of the extended TECDOC as an important and timely taken step and suggested facilitating the development of the document with the goal to publish it in 2014-early 2015. The paper provides further details on the attributes of Fire PSA introduced in the new version of TECDOC-1511.

1. Background

During the years IAEA is developing various safety guides and other technical publications to support nuclear power plant safety in the Member States. In the area of PSA, IAEA Safety Guides SSG-3 “Development and Application of Level-1 PSA” and SSG-4 “Development and Application of Level-2 PSA” [2,3], which were published in 2010 provide high level recommendations for the development and use of PSA for internal initiating events, internal and external hazards at all operational modes.

There are many other publications providing guidance for particular PSA aspects tasks, but one of the most recent and most comprehensive is the TECDOC-1511 “Determining the Quality of Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants” [1]. It provides an approach and detailed guidance for achieving the technical quality of PSA needed to support various PSA applications. The structure of the document includes PSA Elements, Tasks, and

Attributes. This document was published in 2006, so it predates the Safety Guides on PSA [2,3]. While it is, in retrospect, consistent with the Safety Guide, it only covers internal events at full power. Since it has proven to be very useful for the Member States, it was determined that it should be expanded to cover all of the areas addressed in the Safety Guide. Therefore, in 2011 IAEA initiated a project to expand the TECDOC to cover the additional areas.

A core project team was assembled consisting of the IAEA lead and several team members. Over the period from then until December 2013, the core team developed a draft material covering additional PSA areas: internal and external hazards, low power and shutdown models.

An important feature of the document is that the new material was integrated into the existing structure of the TECDOC instead of creating new sections for each hazard. This is different to the approach taken by ASME and ANS in their project on developing the PRA Standard(see for example [4]), where separate documents are dedicated to particular hazards (e.g. the ASME/ANS PRA Standard has Part 4 for fire, Part 5 for seismic, etc.). The TECDOC puts the additional material unique to each hazard into the existing technical elements, therefore avoiding the "back-reference" approach and avoiding duplication of requirements in different parts. New "technical elements" were only added for (1) defining plant operating states and (2) hazard selection, hazard screening, and the definition and quantification of hazard events.

In December 2013 the Technical Meeting (TM) held in Vienna with participation of nineteen representatives of fifteen MSs and one international organization (EC/JRC).

The TM reviewed the draft TECDOC and provided recommendations for further elaboration of the document. It is important the TM appreciates the work done so far and recommends facilitating the work on the document to make it ready for publications in 2014 – early 2015. Work continues on the document, with the goal to produce a final draft for consideration by the Member States in the end of 2014.

2. Summary of Major Changes to the TECDOC-1511

As mentioned above, the IAEA has taken an approach, which is intended to avoid repetitions, “back-references” and even sometimes maybe confusions caused by inclusion of potentially contradicting requirements if the approach of “separate” parts would have been taken. The approach for the extended TECDOC-1511 incorporates the new material primarily into the existing technical elements.

Tables 1 and 2 below show the structure of the revised document and briefly summarize the changes to the original TECDOC aimed to extend it to other than internal initiating events and full-power modes of operations.

Table 1. **Major changes to the TECDOC-1511 (Sections 1- 5)**

Section/subsection of the revised TECDOC	Summary of the information provided	Summary of changes
1. Introduction	Background, objectives of the report, quality of a PSA for an application, scope and structure of the report, applicability of the report	Minor changes reflecting increased scope of the report
2. Overview of PSA applications	Overview of PSA application categories, discussion on the use of PSA results for PSA applications and typical risk metrics used in decision making	New categories of PSA applications and new PSA applications are discussed

Section/subsection of the revised TECDOC	Summary of the information provided	Summary of changes
3. Procedure to achieve quality in PSA applications	Description of main PSA elements, coding scheme used for attributes identifiers, connection to IAEA PSA guidelines and the procedure on how to use the TECDOC	New PSA elements are added: Plant Operational States Analysis (OS) and Hazards Events Analysis (HE)
4. PSA Element 'OS': Plant Operational States Analysis	Two subsections are included: 4.1. Main Objectives 4.2. POS Analysis Tasks and Their Attributes	Completely new sections and sub-sections (i.e. this line and ones below); however the overall structure of original Sections 4-10 of TECDOC-1511 is maintained
4.1. Main Objectives	Addresses the scope expanded to cover all operating modes and defines the objectives of POS analysis - to identify reasonably complete set of unique reactor and plant conditions (i.e. POSs) so that no significant contributor to core damage is omitted. In this context, the full-power operation mode is one of the POS that should be considered in the all-operational-modes PSA. The subsection describes important aspects of the POS analysis.	
4.2. POS Analysis Tasks and Their Attributes	The main tasks for the PSA element 'POS Analysis' and general attributes associated with each one: OS-A - Identification of POSs OS-B - POS Grouping OS-C - Estimation of POS average durations OS-D – Documentation	
5. PSA Element 'HE': Hazards Events Analysis	Two subsections are included: 5.1. Main Objectives 5.2. Hazard Events Analysis Tasks and Their Attributes	
5.1. Main Objectives	Addresses the scope of expanding to cover all hazards and defines the main objectives of the hazard events analysis as follows: To identify a reasonably complete set of the hazards that have the potential to cause initiating events (i.e., interrupt normal plant operation and that require successful mitigation to prevent core damage), so that no significant contributor to core damage is omitted. To address the treatment of correlated hazard events; To provide estimates for the frequencies of the hazard events using information available and associated estimation techniques. The subsection describes important aspects of the HE analysis.	
5.2. Hazard Events Analysis Tasks and Their Attributes	The main tasks for the PSA element 'Hazard Events Analysis' and general attributes associated with each one. HE-A: Identification of Potential Hazards - A list of potential hazards is defined that is as complete as possible. HE-B: Hazard Screening and Final Hazards List Identification -The list of potential hazards is screened in order to eliminate hazards that do not have a significant impact on risk. HE-C: Definition of Hazard Events for Most Hazards - The specific hazard events that are to be considered in	

Section/subsection of the revised TECDOC	Summary of the information provided	Summary of changes
	the PSA are defined in terms of the parameters and characteristics that will best support the determination of the response of the SSCs to occurrence of the hazard. HE-D: Definition of Hazard Events for Internal Fires - A list of hazard events for internal fire (internal fire events) is defined which is as complete as possible. HE-E: Definition of Hazard Events for Internal Floods - A list of hazard events for internal flood (internal flood events) is defined which is as complete as possible. HE-F: Definition of Hazard Events for Seismic - The approach to hazard events for seismic is established and the hazard events are defined. HE-G: Frequency of Hazard Events for Most Hazards - The specific hazard events that have been defined are analysed to determine their frequency. HE-H: Frequency of Hazard Events for Internal Fires - Frequency of fire hazard events in individual fire compartments is assessed based on relevant generic industry and plant-specific evidence. HE-I: Frequency of Hazard Events for Internal Floods - Frequency of flood hazard events in individual flooding area is assessed based on relevant generic industry and plant-specific evidence. HE-J: Frequency of Hazard Events for Seismic - The hazard events for seismic that have been defined are analyzed to determine their frequency. HE-K: Documentation	

Table 2. **Major changes to the TECDOC-1511 (Sections 6- 13)**

Section/subsection of the revised TECDOC	Summary of the information provided	Summary of changes
6. PSA element 'IE': initiating events analysis	Two subsections are included: 6.1. Main Objectives 6.2. Initiating Events Analysis Tasks and Their Attributes	The inclusion of other plant operating states and other hazards does not require significant changes to the attributes under this element. A few new attributes are required in order to address the issue of tying hazard events to initiating events, and identifying initiating events that may be unique to specific hazards.
7. PSA element 'AS': Accident sequence analysis	Two subsections are included: 7.1. Main Objectives 7.2. Accident Sequence Analysis Tasks and Their Attributes	The integration of new material only required editing text to cover consideration of POS.
8. PSA element 'SC': Success criteria formulation and supporting analysis	Two subsections are included: 6.1. Main objective 6.2. Success criteria formulation and supporting analysis tasks and their attributes	Same as above

Section/subsection of the revised TECDOC	**Summary of the information provided**	**Summary of changes**
9. PSA element 'SY': Systems analysis	Two subsections are included: 9.1 Main objectives 9.2. Systems analysis tasks and their attributes	The systems analysis part has relatively more changes in order to incorporate the new material. It is necessary here to address the inclusion of SSCs and failure modes that would not be treated in the attributes as they currently are focused on internal initiating events. For example, the fire equipment selection attributes are addressed here, as are certain aspects of the development of the seismic equipment list.
10. PSA element 'HR': Human reliability analysis	Two subsections are included: 8.1. Main objectives 8.2. Human reliability analysis tasks and their attributes	There are various considerations of human reliability that are addressed for the integration of LPSD attributes into this section, of particular note being the treatment of human-caused initiating events due to the greater level of human interaction with the plant during the changes in POS involved in shutdown and start-up and the extensive level of maintenance activities (including refuelling) that can take place. This requires some significant changes. Post-initiator HRA attributes are mostly handled by re-wording the existing attributes, but there are also new added attributes to fully cover the topic.
9. PSA element 'DA': Data analysis	Two subsections are included: 9.1. Main objectives 9.2. Data analysis tasks and their attributes	-
9.1. Main objectives	Data analysis becomes the determination of the failure probability of the systems, structures, and components considered in the model.	The additions to the data analysis element are more extensive, because it was a broad view taken of what are the objectives of the data analysis element.
9.2. Data analysis tasks and their attributes	With the broader objective of the data analysis element, it can now encompass fragility analysis, and also the circuit analysis required for fire PSA (which could be considered as a type of fragility analysis, but is best considered as a unique type of "data analysis"). New tasks area added, e.g.: DA-H - When the primary mechanism is induced by a physical phenomenon, as is the case for internal and external hazard events, the failure probabilities are determined by mechanistic analysis, referred to as fragility analysis.	With this approach, the expanded types of data analysis cannot as easily be integrated into the guidelines with editorial changes to the existing attributes or the addition of a few new attributes. Instead, new tasks need to be added.
10. PSA element 'DF': dependent failures analysis	Two subsections are included: 10.1. Main objectives 10.2. Dependent failure analysis tasks and their attributes	The integration of new material required modification of the attributes related to such things as seismic failure correlation, etc.

Section/subsection of the revised TECDOC	Summary of the information provided	Summary of changes
11. PSA element 'MQ': model integration and CDF quantification	Two subsections are included: 11.1. Main objectives 11.2. Model integration and CDF quantification tasks and their attributes	For the most part, the existing attributes for quantification are adequate for the expanded scope. It is only necessary to address that each POS should be quantified individually and also to address the need to convolute hazard and fragility when appropriate. Also an attribute for consideration of low success probabilities (which is not an important issue for internal events PSA) needs to be added.
12. PSA element 'RI': results analysis and interpretation	Two subsections are included: 12.1. Main objectives 12.2. Results analysis and interpretation tasks and their attributes	The key issue that needs to be addressed here is aggregation; what is the appropriate way to speak of the total risk from all hazards and modes given the different levels of conservatism that may exist.
13. Determination of special attributes for PSA applications	The expanded scope covers areas that will have applications beyond those that were included only for internal events, and so special attributes needed for those applications will be needed.	The addition of LPSD and internal/external hazards will expand the scope even of the current internal events applications. This section will include the additional special attributes required.

3. Summary of the Changes Related to a Fire PSA in TECDOC-1511

As it is shown in Tables 1 and 2 most of the PSA elements are modified and new tasks are added in order to address new areas of applications of the extended TECDOC-1511.

These changes address also the inclusion of internal fires in the scope of the document. There are two major changes in the document that allow covering internal fires:

1) Adding specific tasks
    a. Task HE-D 'Definition of Internal Fire Hazard Event'
    b. Task HE-H 'Frequency of Internal Fire Hazard Event'.
2) Adding specific attributes related to an internal fire PSA

Table 3 below provides an extraction from Table 5.2-D - Attributes for HE Analysis: Task HE-D 'Definition of Internal Fire Hazard Event'. The attributes provided in Table 5.2-D are aimed to ensure that Internal Fire Hazard events are properly identified for all operational modes and none of the fire hazard events is screened out without a proper justification.

Table 3. **Example of attributes for the task 'Definition of Internal Fire Hazard Event'**

Task / GA	Characterization of Task/General Attributes. *Identifier and Description of Special Attributes (in Italics)*	Rationale/Comments/Examples for General Attributes and *Special Attributes (in Italics)*
HE-D	A list of hazard events for internal fire hazard events is defined which is as complete as possible.	DEFINITION: Internal Fire Hazard Event is an event brought about by the occurrence of a fire within the plant boundary, and which directly or indirectly damages components and causes initiating events and may further cause safety system failures or operator errors that may lead to

Task / GA	**Characterization of Task/General Attributes.** ***Identifier and Description of Special Attributes (in Italics)***	**Rationale/Comments/Examples for General Attributes and** ***Special Attributes (in Italics)***
		core damage or large early release. Internal fire hazard events are generally defined in terms of location, ignition source, heat release rate profile, and extent of propagation.
HE-D01	All modes of operation with power operation and shutdown modes are considered. Fire Hazard Events applicable to specific operational modes are identified.	RATIONALE: Some fire hazard events can only occur during specific operation mode, may be more likely, or have a higher HRR during specific modes of operation; and this may lead to the need of operation mode specific fire hazard event analysis. EXAMPLE: Welding activities, which may be a significant cause for fires are usually performed during shutdown mode of operation. Therefore, the conditions for the fire hazard analysis should consider shutdown specific factors, like disabled fire detection, the availability of larger combustible materials, or different fire compartment definition. Example: See the NRC NUREG/CR-7114 for how fires may change during shutdown. In general, we see the following major impacts from the Fire Hazard Event assessment: 1) some fires are much more likely during shutdown, 2) fires may be much larger in HRR, and 3) plant partitioning affecting the extent of propagation can be much larger. Lesser impacts would be inoperability of fire detection and suppression, which can be maintained at any time for most systems.
HE-D02	Plant specific information required for the internal fire hazard event definition and analysis is collected to support the fire hazard event definitions: a) Cable routes of the plant, including raceways, conduits, trays and barriers; b) Equipment layout in different rooms; c) Equipment and cable failure modes potentially induced by the fire; d) Fire damage criteria; e) Data on fire events; f) Human actions in the event of a fire and human error probabilities; g) Fire loads in compartments; h) Fire protection procedures.	RATIONALE: Most of the information supporting the internal fire hazard event definitions is plant specific (or equipment specific) and cannot be taken from generic information sources. COMMENT: Not all of this information will be needed for the fire hazard definition; however, the data collection should cover all these items to support the whole PSA. Comment: Equipment failure modes potentially induced by fire may include failure to operate, fail-as-is, or spurious operation. NOTE: In the case of PSA for basic design stage of the plant this attribute is not applicable. Apply special attribute HE-D02-S1 instead.

Task / GA	**Characterization of Task/General Attributes.** ***Identifier and Description of Special Attributes (in Italics)***	**Rationale/Comments/Examples for General Attributes and** ***Special Attributes (in Italics)***
	Special Attribute HE-D02-S1: *In PSA for basic design stage the following information is applied for fire hazard event definition:* *a) Any information available at design stage from the list in HE-D02* *b) For the missing information realistic assumptions are made, and the assumptions are based on the experience gained in sister or similar plants if they exist.* *OR* *The fire PSA is based on the assessment of the availability of Safe Shutdown Equipment if already declared at the design phase.*	*COMMENT: In the design stage of a plant the information needed to support the internal fire PSA is not complete or simply not yet available, therefore the needed information should be somehow "created" making assumptions.*
HE-D03	Plant walkdowns are performed to verify the accuracy of information obtained from drawings and other sources of plant information, and to obtain necessary information on spatial interactions for analysis of fire propagation from each potential fire source and damage to potential equipment or cables.	RATIONALE: Walkdowns are necessary to confirm any plant partitioning credited in the definition of fire compartments, as well as to determine the spatial interactions such as the distance between the source and the targets (equipment or cables), location of fire detection/suppression, the potential ignition of secondary combustibles, and other fire hazard event specific information. COMMENT: While performing a PSA for a plant in the design stage, walkdowns are effective only by the end of plant construction.
----	------	------
HE-D09	The effectiveness of raceway fire wraps other passive fire barrier elements, or active fire barrier elements credited in the analysis of fire scenarios are evaluated.	Note: Damage to a credited fire barrier due to a high energy event should be considered in fire scenarios, where applicable.
HE-D10	For multi-unit sites with shared systems or structures, potential multi-unit impacts of internal fires on SSCs and plant initiating events are considered and analyzed. The number and combination of reactor units and related operational modes affected by the fire are identified.	

The second table 5.2-H - Attributes for HE Analysis: Task HE-H 'Frequency of Internal Fire Hazard Event' provides attributes that ensure proper frequency assessment of fire events in all operational modes.

The other attributes related to internal fire PSA are included in various places of the extended TECDOC. Some examples of such General and Special attributes are provided in Table 4 below in accordance with major PSA elements.

Table 4. **Example of fire-related attributes for selected specific PSA tasks**

Identifier of the Attribute	**Description of Attributes *(for Special Attributes in Italics)***	**Rationale/Comments/Examples for General Attributes and *Special Attributes (in Italics)***
Task: Initiating events analysis		
IE-A18	The impact analysis includes the results from the circuit analysis performed in the frame of the fire PSA in order to identify initiating events with spurious actuations.	RATIONALE: The spurious actuations of different system or equipment protections may lead to PSA initiating events. NOTE: This attribute is applicable mainly to the PSA for internal fires; however, certain considerations should be given to internal floods.
IE-A120	Internal fire in the main control room is included in the PSA including initiating events from loss of function and fires causing a control room abandonment.	RATIONALE: The special effects of a fire in the main control room have to be included into the fire PSA taking into account the specific features associated with this location.
Special Attribute IE-G12-S	*The detailed impact analysis for fire initiating events includes deterministic fire propagation calculations using qualified computer codes in order to reduce the unnecessary conservatism of the associated assumptions.*	*COMMENT: The reduction of the conservatism in the fire PSA enables specific applications (like risk informed fire protection)*
Task: Accident sequence analysis		
AS-C17	In the PSA for fires special event sequence model is developed for the fire in the control room and the model takes into account the following special effects: - widespread effect of a fire in the main control room across all safety systems, - the potential for spurious actuation of systems and, - the impact of fire in the main control room on operator actions. ….	RATIONALE: In the PSA for internal initiating event usually there is no such event tree model that can be applied to model the fire in the control room. It is an initiating event special to fire hazards, therefore a special event sequence model should be developed.
Task: System Analyses		
SY-B11	If hazard-induced fires or floods are identified as part of the dependency and fragility analyses, their effects are added to the model.	-
SY-C18	In the PSA for hazards the basic events representing the SSCs belonging to the set of equipment affected by the hazard are identified, and their probability is adjusted, or the relevant logical switches are set to the appropriate status for considering the effect of the hazard.	EXAMPLES: 1) In seismic PSA the probabilities of the basic events affected by the earthquake are calculated using the relevant fragility curves. 2) The basic events representing fire affected equipment considered to be disabled by the fire are either set to logical TRUE status, or a suitable logical switch in the fault tree model is set to logical TRUE.

Identifier of the Attribute	**Description of Attributes *(for Special Attributes in Italics)***	**Rationale/Comments/Examples for General Attributes and *Special Attributes (in Italics)***
Task: Human reliability analyses		
Special Attribute HR-E01-S3	*Actions taken outside the control room in response to procedural direction or control room abandonment are identified for specific accident sequences related to specific hazards (e.g., fire, flooding, seismic, etc.).*	*RATIONALE: Control Room abandonment may be required due to the control room becoming un-inhabitable, or as a result of loss of functions needed to protect the core.*
Special Attribute HR-E04-S1	*Identify any new undesired operator actions in response to fire-induced spurious indications or alarms.*	*Comment: Undesired operator actions are in response to procedural compliance following a spurious alarm. These actions may result in stopping equipment needed in the FPSA, but may be recoverable.*
HR-H05	Recovery actions that cannot be performed due to the impact of the hazard of certain severity are removed from the PSA model or probabilities of failure whilst performing the action are adjusted.	EXAMPLE: If there is no access to the fire affected room, the field operator cannot perform manual closing of the valve located in that room, which was considered in the PSA for internal initiating events as a recovery of the failure of automatic closing of the same valve due to loss of electric supply of the valve.
Task: Data Analysis		
DA-E03	A mean value of, and a statistical representation of the uncertainty intervals for the parameter estimates is provided.	COMMENT: Acceptable systematic methods include for instance: Bayesian updating or expert judgment. COMMENT: Uncertainty estimates are required for parameters associated with both internal and external events, including for example fire non-suppression estimates, severity factors, conditional events following an external event, etc.
Special Attribute DA-H01-S1	*For Fire PSA, the fire-induced circuit failure probabilities are included in the model. The probabilities are based on available generic data, using plant-specific circuit analysis.*	
Task: Dependency analysis		
DF-C02	Physical dependencies resulting from internal and external hazard events are considered. This is done during the plant walkdown for the fragility analysis. Particular attention is paid to the potential for hazard-induced fires and floods in addition to the items discussed in DF-C01.	COMMENT: See also the related attribute DA-G02.
Task: Model quantification and integration		
MQ-B03	Develop the hazard specific PSA model so that systems and equipment that were included in the internal-events PSA but are not credited in the hazard-specific PSA and that are potentially vulnerable to hazard failure, are failed in the *worst* possible failure mode.	COMMENT: The final PSA model should not include any significant equipment failed per this requirement, unless it can be demonstrated that additional modelling will not impact the results. NOTE: for Fire PSA, the worst possible failure mode can include fire-induced spurious operation.

Identifier of the Attribute	**Description of Attributes *(for Special Attributes in Italics)***	**Rationale/Comments/Examples for General Attributes and *Special Attributes (in Italics)***
Task: Results interpretation		
RI-A01	Significant contributors to CDF are identified. The contributors are, in increasing level of resolution: - Hazard events - Plant operational states - Initiating events - … - Fire compartments or scenarios (in fire PSA) - Flooding areas (in flooding PSA) ….	

4. Conclusions

In 2011, IAEA initiated a project to expand its TECDOC on PSA Quality to cover Low Power and Shutdown modes, internal and external hazards. IAEA decided to take an innovative approach to this by integrating the new quality attributes into the current structure of the TECDOC, as opposed to the ASME/ANS approach of creating different parts of the TECDOC for the attributes for each hazard and modes of operation. In the work that has been completed this far, this approach has been shown to be workable and will eliminate the need to "back-reference" to a separate internal events part that has proven so unwieldy in the use of the ASME/ANS standard, and will eliminate the need to repeat common material in each part. It is believed that this will make the use of these guidelines easier for both the practitioners and the reviewers.

Significant part of the document is aimed at addressing internal fire hazards, both originated from fire sources inside the plant and caused by other hazards (e.g. seismically induced fires).

There are issues that are not currently addressed in the document and need further elaboration. These include:

- Internal fires caused by other hazards (seismic, internal and external floods, etc.)
- Potential wider fire propagation in shutdown conditions
- Fire-induced floods, etc.

Some of these areas are currently beyond state of the art; however, the document will suggest certain attributes to cover these aspects.

This project is still underway, and it is hoped that a final document can be completed by the end of 2014.

References

[1] IAEA, Determining the Quality of Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants, TECDOC-1511, International Atomic Energy Agency, Vienna, Austria, (2006)

[2] IAEA, Development and Application of Level-1 PSA, SSG-3, International Atomic Energy Agency, Vienna, Austria, (2011)

[3] IAEA, Development and Application of Level-2 PSA, SSG-4, International Atomic Energy Agency, Vienna, Austria, (2011)

[4] ASME/ANS RA-Sa-2009 (Addenda to ASME/ANS RA-S-2008), "Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009

Focus on the experimental investigations in support of fire PSA concerning the French 1300 MWe Nuclear Power Plants

J. Espargillière, T. Vinot, L. Audouin
Institute for Radiological Protection and Nuclear Safety (IRSN), France

Abstract

As a part of the on-going third Periodic Safety Review (PSR) of the French 1300 MWe Nuclear Power Plants (NPPs), IRSN (French TSO) develops a simplified fire level 1 Probabilistic Safety Assessment (fire PSA) for NPPs in order to assess the assumptions and the results of the NPPs licensee's (EDF) fire PSA studies. Fire PSA performed by IRSN is a part of the IRSN in-house NPP level 1 PSA for the internal events.

IRSN fire PSA focuses on the most critical safety equipment during fire scenarios in terms of fire-related risks. Indeed, for each fire source in each room, the impact of a fire is assessed by carrying out fire simulations with SYLVIA fire code based on two-zones modelling (0). The numerical results allow identifying precisely all the important safety equipment that are potentially damaged by fire stress and providing the associated failure time. From this study, the core meltdown frequency is then evaluated using the results of these fire simulations.

To perform properly these fire simulations, IRSN needs to determine fire properties and failure criteria of electrical safety equipment by means of experimental tests to establish:

- the fire source characteristics (heat release rate, fire growth, combustion products…) based on open and confined fire tests representative of fire scenarios in NPPs;
- the malfunction criteria of electrical safety equipment due to smoke and heat stresses coming from compartment fire. The fire tests conducted in IRSN facility showed that the proper assessment of smoke impact on equipment failure depends on a coupling effect of both soot concentration and gas temperature surrounding the safety equipment of interest. Consequently, IRSN has developed an original experimental apparatus in order to design the first small-scale test method for determining the malfunction of electrical equipment by taking into account both soot concentration and gas temperature.

This paper presents the key elements of R&D studies in support of the fire PSA of the French 1300 MWe NPPs. This technical investigation described the experimental tests (cabinet fire, electrical malfunction of safety equipment) performed at IRSN, the different assumptions for the fire scenarios and the fire modelling used in SYLVIA code (two-zones modelling) to perform calculations in fire PSA studies. The main assumptions proposed in this paper (how to model a fire for a row of eleven electrical cabinets, what fire effects considered for electrical malfunction) is discussed in detail.

1. Introduction

Over several years now, the French nuclear fire safety regulation (0, 0) has turn from prescriptive requirements to objective requirements. Consequently, the Fire Safety Analysis (FSA) focuses now on the compliance of fire effects with performance criteria for fire protection measures. These performance criteria are mainly related to the vulnerability of targets (for instance, electric/electronic equipment and control/power cables) within Nuclear Power Plants (NPP) in order to avoid accidental sequences leading potentially to core melting of nuclear reactors. To reach this objective in its FSA, the NPP's licensee has to define thoroughly the major safety functions of its nuclear facility and has to ensure that nuclear safety objectives are fulfilled in the case of accidental fire. A key step in this analysis process is the identification of the most critical targets associated to the major safety functions. These critical targets could include structural elements, all types of nuclear systems and various components important to nuclear safety. Further, they could also concern employees who have to ensure safety operations for NPPs. To perform the fire-risk analysis, the fire features related to nuclear facility of interest needs to be determined for all operating conditions as: geometry of rooms, combustible loads stored in compartments, potential ignition sources, characteristics of ventilation network, fire protection/detection systems available within the nuclear installation, and so on. From this database, the fire PSA studies are then led by using the lessons learned from the French fire PSA developed initially in the 90's for 900 MWe NPPs, which is in accordance with the international methodology proposed by EPRI/NRC in 2005 (0). Indeed, the fire PSA methodology (0) consists of three successive steps as: (1) the critical compartments selection, (2) the quantification of fire scenario (effects, frequency), and (3) the quantification of frequency of core damage due to selected fire scenarios. The outcomes from this fire PSA studies provide valuable knowledge on NPPs' fire hazards to identify the most critical equipment and fire scenarios. Indeed, they showed that critical fire scenarios are mainly linked with fire hazards in the electrical buildings. Consequently, IRSN developed a simplified fire level 1 PSA especially for NPP electrical buildings, as a part of the on-going third periodic safety review of the French 1300 MWe NPPs and a part of the IRSN in-house NPP level 1 PSA for the internal events. The major objective of this fire PSA was to assess the assumptions and the results proposed in the EDF fire PSA studies (0). Thus, IRSN fire PSA focuses on the most critical safety equipment during fire scenarios in terms of fire-related risks to reach the core melting.

In order to assess the fire consequences for each fire source and each room of interest, a series of fire simulations was carried out by means of SYLVIA fire code developed by IRSN (0). This fire code is based on two-zones modelling (0, 0) and can calculate the smoke and heat transfers from fire source to other compartments, ventilation network and targets. The results obtained from theses fire scenarios allowed identifying accurately all the important safety equipment that could be potentially damaged by fire consequences (including thermal and soot stresses) and also providing the failure time related with target damages (electrical devices and cables). From these works, the quantification of fire scenario frequency is possible and then the assessment the core meltdown frequency (0) for the fire PSA studies in the French 1300 MWe NPPs.

To perform properly these fire simulations and predict the target damages, IRSN needs to determine fire properties and failure criteria of electrical safety equipment by means of experimental tests to establish:

- the fire source characteristics (heat release rate, fire growth, mass loss rate, combustion products…) based on open and confined fire tests representative of fire scenarios in NPPs;

- the malfunction criteria of electrical safety equipment (relays, electronic boards, switchgear…) due to smoke and heat stresses generated by compartment fire tests. Indeed, the fire tests conducted in IRSN facility (0) showed that the proper assessment on equipment failure depends on a coupling effect of both soot concentration and gas temperature surrounding the safety equipment of interest. This outcomes will be discussed further in the paper;
- the malfunction criteria of power/control cables based on the critical value of temperature inside the cables, which is obtained from experimental tests (0, 0). Nevertheless, this point will be not considered in this paper.

After this introduction, some experimental tests performed at IRSN were presented concerning the determinations of characteristics of electrical cabinet fire and malfunction criteria of electrical equipment. From these experimental outcomes and a dataset (complex geometry, ventilation network…) describing the NPPs of interest in fire PSA study, the fire modelling is proposed and the key assumptions needed for the calculations by means of SYLVIA fire code are detailed. A discussion is proposed about the damage criteria of electrical equipment before concluding this paper.

2. Experimental Tests for Determining the Characteristics of Electrical Cabinet Fire

In order to simulate properly a fire scenario with a fire code, the first stage consists in the determination of the fuel source fire properties. Consequently, fire tests concerning real electrical cabinets (0, 0 and 0) were performed in an open atmosphere under a large-scale calorimeter, and aimed to characterize the cabinet fire properties in terms of:

- heat release rate;
- incident radiant heat flux in front of the cabinet;
- combustion products (O_2, CO_2, CO, soot).

Two configurations were mainly investigated:

- some open-door cabinets (see Figure 1) allowing the fire to freely growth along wires and components inside cabinet in free atmosphere;
- some closed-door cabinets (see Figure 2) with two square openings: one on the top part and the other on the bottom part of door.

The real electrical cabinet tested was an electro-technical cabinet with two modules. This cabinet with steel walls measured 2 m in high, 1.2 m in width and 0.6 m in depth. The main electrical components were transformers, circuit breakers, cables, trunkings, relays, terminal blocks, contact switches and contactors. The total combustible load was assessed at 44 kg and consisted of about 32% of Polyethylene Vinyl Acetate (PVA), 30% of PolyVinyl Chloride (PVC), 26% of PolyAmide (PA), 9% of PolyEthylene (PE) and 3% of other compounds. The cabinets were not electrically supplied. In all experiments, the fuel materials were ignited by a linear propane gas burner located at the bottom of cabinet. More technical details about these fire tests are available in (0, 0 and 0).

For real electrical open-door cabinets, a significant quantity of smoke appeared just after ignition and the flame spread slowly from the bottom to the top, all along the electrical components. A few minutes later, the fire were fully developed on electrical components (Figure 1) leading to a powerful fire (Figure 3). In this configuration, all the combustibles were burnt.

Just after ignition for real open-door cabinets, smoke was observed exiting from the upper ventilation openings as showed in Figure 2. Moreover, flames could also appear through these openings (0). Sometimes, puffs of smoke could exit at the lower ventilation openings (0). Depending on material nature, combustible load and opening sizes, the fire could be quickly extinguished by lack of oxygen (0, 0) leading to a weaker amount of material pyrolysis in comparison with real open-door cabinet.

Figure 1. **Fire test of real open-door cabinet under large-scale calorimeter**

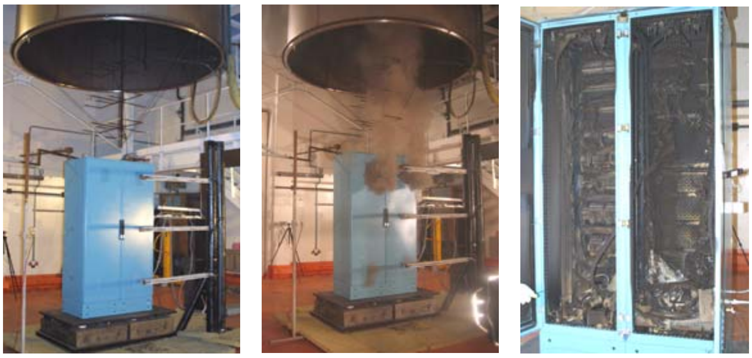


Figure 2. **Fire test of real closed-door cabinet under large-scale calorimeter**

From all fire experiments carried out in IRSN experimental laboratory (0 to 0), the fire tests showing the highest Heat Release Rate (HRR) for open and closed-door cabinets have been considered in this study and are shown in Figure 3. Furthermore, the HRR curve concerning closed-door cabinet (CA02 test in 0) is very similar with curve of real cabinet fire performed previously by Mangs (0, 0).

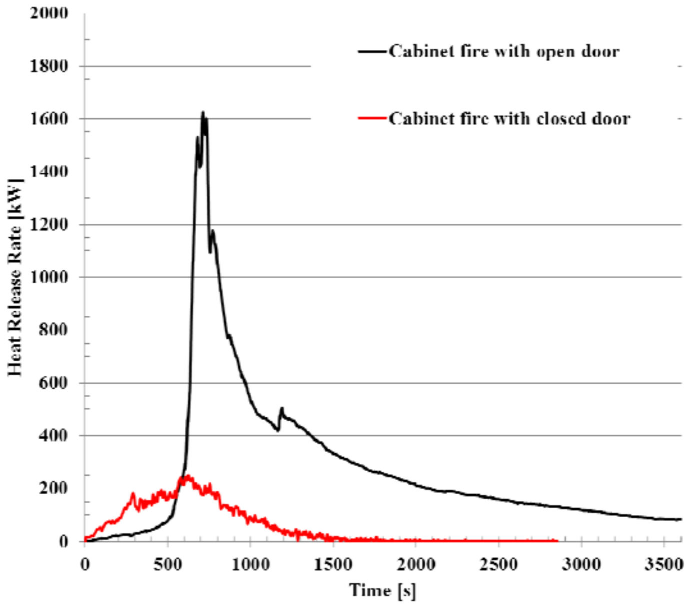


Figure 3. **HRR of open and closed-door cabinets**

3. Experimental Tests for Determining the Malfunction Criteria of Electrical Equipment

In a first stage, analytical tests for determining thermal damage on equipment of interest were carried out for several electric equipment available on French NPPs: two relays (VIGIRACK and MICOM) and one circuit breaker (UNELEC). These electrical components were exposed to a convective thermal stress in a furnace ranging from ambient (i.e. about 20°C) to 300°C in temperature and from 0.01 to 1 $m.s^{-1}$ in air velocity around the equipment. Similar tests were performed on three relays (Agastat GPI and General Electric HMA) by Sandia National Laboratories (0), on which was applied thermal stress in an environmental chamber by step increases of temperature. The thermal failures of theses relays were observed for temperatures ranging from 150°C to above 350°C. Concerning thermal failure tests from IRSN, electrical malfunction of VIGIRACK relay (0) occurred when the ambient temperature reached about 165°C (including repeatability tests). This result seems to be consistent with previous SNL study.

In a second stage, this same VIGIRACK relay was tested in real fire conditions during a series of large-scale fire tests performed in DIVA facility (0, 0, 0). The experimental conditions consisted in a real open-door cabinet (see previous paragraph and Figure 1) as fire source in fire room 1 (120 m^3 in volume). This fire room (Figure 4) was surrounded by three adjacent compartments (two rooms and one corridor) and was separated from the room 2 by a firebreak door and was linked with corridor by means of a circular calibrated leak. The rooms 2 and 3 were also connected with the corridor with similar circular leaks. The relay was located inside the fire room 1, which was ventilated with a renewal rate of about 2 h^{-1}.

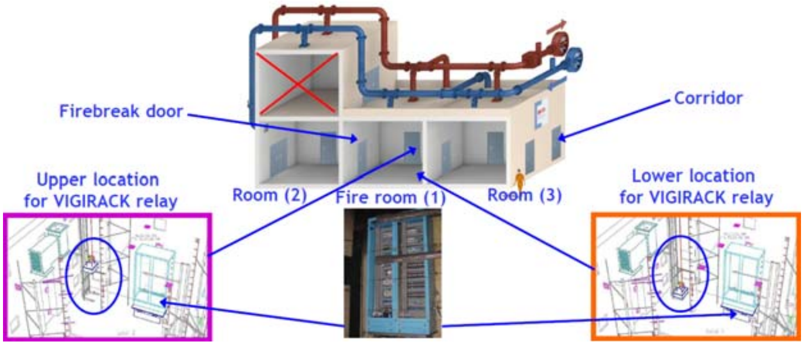


Figure 4. **Fire scenario for studying malfunction of electrical equipment (0, 0, 0)**

In the DIVA facility, various ventilation control strategies and locations of the relay were tested, as described in Table 4 hereafter.

Table 1: **Location, strategy for ventilation and damage for VIGIRACK relay**

Test n°	Location of relay in fire room	Ventilation control strategy	Relay malfunction observed?
1	Corner of fire room Height: 1.80 m from floor	-	Yes
2	Corner of fire room Height: 0.55 m from floor	Closure of inlet damper (3min)	No
3	Corner of fire room Height: 0.55 m from floor	Closure of inlet (3min) and outlet (8min) dampers	No
4	Corner of fire room Height: 1.80 m from floor	Closure of inlet and outlet dampers (8min)	Yes

As an unexpected result (0), some electrical malfunctions of VIGIRACK relays were observed at ambient temperature (measured nearby the relay), well below the malfunction temperature previously measured in the analytical furnace (i.e. 165°C). After a thorough analysis of experimental data, a coupling effect of both temperature (T) and soot concentration (Cs) on this equipment is highly suspected. Indeed, malfunctions of relay were obtained for soot concentration higher than $1.5 g.m^{-3}$ and ambient temperature over 61°C, as shown in Figure 5 (red dotted line). Conversely, the relay was showed to function well again as soon as the ambient environment decrease below the previous values for both temperature and soot concentration. It should be emphasized that the fire source included about 30% of PVC and thus a significant amount of hydrogen chloride was released during fire tests. But no failure due to corrosive effect was observed during the tests. Furthermore, the VIGIRACK relays were experienced electrically for a full week without finding any new failure. This outcome showed no short-term corrosive effect in our fire tests.

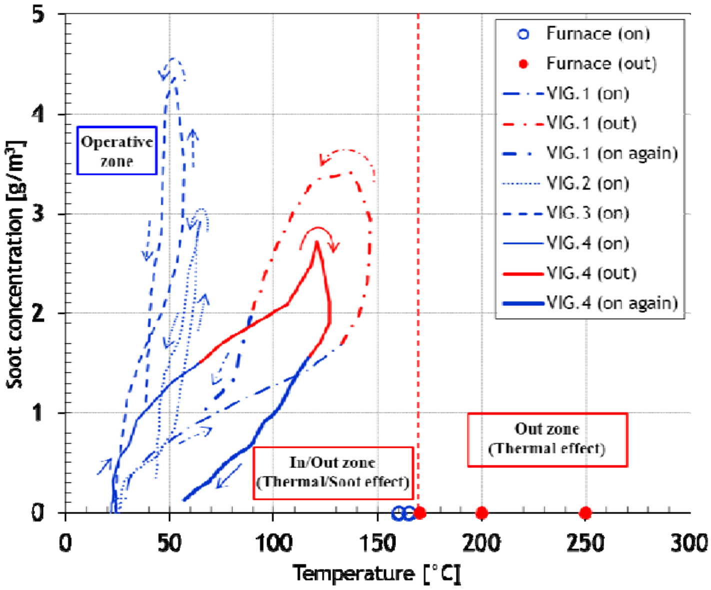


Figure 5. **Time evolution of electrical malfunction of VIGIRACK relay vs (T, Cs) (arrows showing the way of time)**

Consequently, in order to investigate thoroughly the thermal/soot malfunction of electrical equipment important for nuclear safety, IRSN decided to design an original experimental apparatus for determining the malfunction of electrical equipment exposed to both gas temperature and soot concentration. This new furnace is foreseen to provide ambient temperatures ranging from 20°C to 250°C and soot concentration from 0 to 5 g.m^{-3}. This apparatus is now constructed and the first trials to validate it are underway.

3. Fire Simulations for Fire PSA Studies: Numerical Tool, Dataset and Fire Modelling

3.1 Numerical tool

The SYLVIA software system (0), developed by IRSN, is designed to simulate the fire growth and its consequences in an industrial facility featuring a ventilation network. Especially, it allows calculating the development of the fire, the transportation of hot gases and soot, the resuspension and the transportation of aerosols (whether radioactive or not), the clogging of filters, the electrical failure of electrical equipment and the mechanical damage on fire barriers such as firebreak doors and fire dampers. Based on a two-zones modelling, each compartment is divided into two zones of variable volume, in which the thermodynamic properties are uniform (temperature, combustion products…). The ventilation network is modelled using a set of elements, such as ducts, filters, valves, fans, etc. Mass and heat exchange correlations (between zones, flames and walls) supplies the mass and energy balance equations performed in each zone. This software is especially designed to perform a lot of simulations (low computed time per run) that are required as part of probabilistic safety assessments.

3.2 Dataset needed for fire simulations

IRSN and EDF performed a measurement session to collect room data (dimensions, wall composition, equipment positions…) and ventilation network data (air flow-rates, leakages rate through openings, pressure and flow-rates through ducts and ventilation equipment…). From this

this database, a comprehensive NPPs electrical building model was developed in SYLVIA fire code. This model is used to perform simulations for all fire scenarios of interest. Due to its specific design, this model is able to assess accurately the fire development and propagation of hot smoke in NPPs and to take into account the success or failure of safety actions including ventilation management.

3.3 Modelling the fire source: How to model the fire spread from electrical cabinet to adjacent ones?

For this study, the fire source is considered to be a row of 11 electrical cabinets, as shown in Figure 6. The cabinet located in the centre of the row is assumed open-door and to be the initial source of ignition. For other cabinets, their doors are closed. Based on realistic approach, it is assumed that the time evolutions of HRRs for open and closed-door cabinets are those presented previously in paragraph 2 and Figure 3. Moreover, the fire spread from cabinet to cabinet is taken into account by assuming the symmetric fire propagation from centre to the ends of cabinet row. Consequently, the total heat release rate is the sum of HRR for each single cabinet already in fire. The fire spread from a cabinet to the adjacent one is modelled with the recommended approach of NUREG/CR-6850 (0), i.e. assuming fire propagation delayed by 15 minutes (i.e. 900 s). The special cases dealing with the absence of fire spreading between two cabinets are very complex and hard to connect with real fire scenarios. So, it is assumed that the fire spreads in any case (conservative approach).

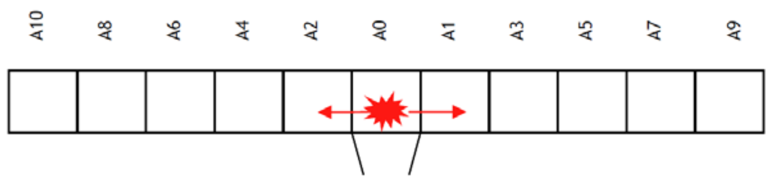


Figure 6. **Symmetric fire spread from open-door electrical cabinet (located in the center of cabinet row) to the adjacent closed-door electrical cabinets**

Based on these previous assumptions, the time-evolutions of heat release rates are presented in Figure 7. During the first 15 min, the HRR curve (blue curve) follows exactly the HRR from experimental open-door cabinet fire (green curve) showing a HRR peak at about 1.6 MW at 710 s. After 900 s, the fire propagates to the two adjacent cabinets (i.e. cabinets A1 and A2 in Figure 6). In the same time, the HRR begins to decrease slowly from about 800 to 700 kW due to both HRR decrease of open-door cabinet and HRR increase from the two closed-door cabinets. Then, 900s yet later, the next cabinets (namely, A3 and A4) ignites and the HRR continues to decrease in nearly the same rate and so on. Of course, this simple approach can be used a priori for any number of cabinets. In order to guarantee a fully conservative approach in fire scenarios, the next assumption considers nearly the same fire growth than the experimental fire source and to keep the HRR constant after its peak as illustrated in Figure 7 (red curve). Indeed, a simplified time-evolution in 3 successive stages (namely, modelling for fire-risk analysis) is then proposed for the heat release in open atmosphere, such as:

- an incubation stage ($0 \le t \le t_i$), in which the fire power growths slowly: HRR = $a_1.t^2$ with $a_1 = 3.7e^{-04}$ kW.s^{-2} and t_i = 600 s (i.e. 10 min);

- a fast spread stage ($t_i \leq t \leq t_{peak}$), in which the rapid fire propagation is considered from low to high HRR (peak): $HRR = a_2.(t - t_i)^2 + a_1.t_i^2$ with $a_2 = 121.42\ kW.s^{-2}$ and $t_{peak} = 710$ s (i.e. nearly 12min);
- then, a steady stage ($t_{peak} \leq t$), in which the fire power is kept constant: HRR = 1.6 MW.

No decay stage for HRR is considered in this study because the numerical outcomes showed that the electrical malfunction of equipements important for nuclear safety appeared very early (less than 2000 s) compared with the potential duration of fire concerning 11 electrical cabinets.

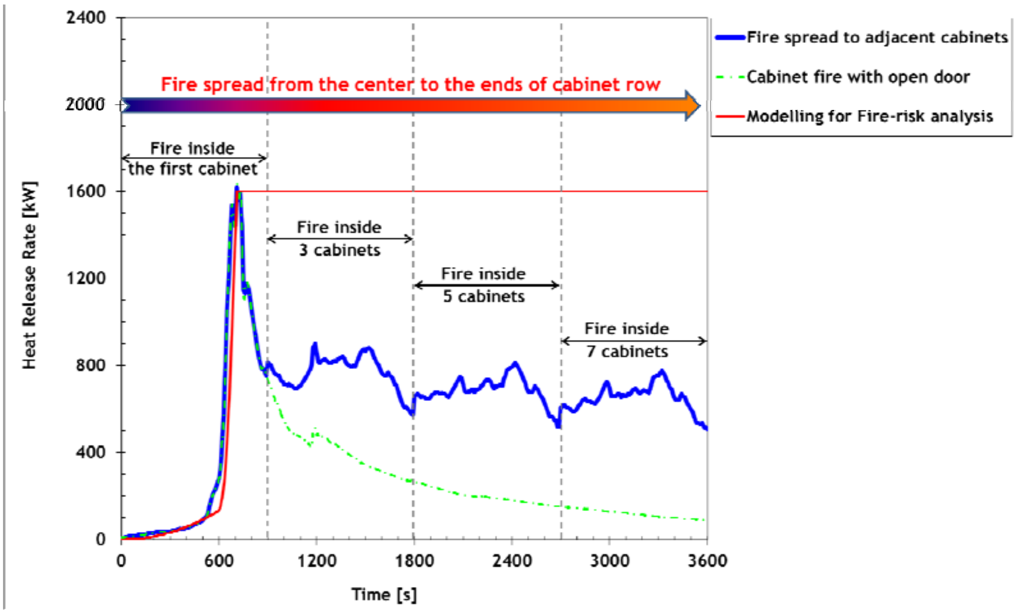


Figure 7. **Modelling of heat release rate (red curve) for 11 electrical cabinet fire (green curve: single open-door cabinet; blue curve: fire spread considering cabinet fire tests)**

In addition, the effect of confinement involves that the fire could extinguish below a given oxygen threshold. Here, an oxygen threshold of 8% in volume is considered. Concerning the simplified combustion reaction defined for fire simulation, the major products of combustion are introduced following the chemical reaction hereafter:

$$Fuel + Y_{O2}\, O_2 \rightarrow Y_{CO2}\, CO_2 + Y_{CO}\, CO + Y_{H2O}\, H_2O + Y_{soot}\, C$$

In this equation, Y_{O2}, Y_{CO2}, Y_{CO}, Y_{H2O} and Y_{soot} are respectively the mass rate of oxygen consumption, the mass rate of production of carbon dioxide, carbon monoxide, water vapour and soot. Soot production rate is chosen as 0.05 g/g. Other chemical product yields (O_2, CO_2, CO, H_2O) are determined from both experimental data (0, 0 and 0) and mass balance.

4. Review of Support Studies of Fire-PSA

From the comprehensive modelling of fire scenarios detailed in the previous paragraph, a set of 77 fire simulations were performed with SYLVIA code and the numerical outcomes were used for the fire PSA studies in the French 1300 MWe NPPs (0). Concerning the electrical malfunction for assessing the most critical equipements for nuclear fire safety (i.e. fire-related risks to reach the core melting), three damage criteria of electrical equipement were considered:

- 1st criterion: as a reference study, a conservative damage criterion based on gas temperature surrounding the electrical equipment higher than 65°C is assumed (0);
- 2nd criterion: from this reference study, a first sensitivity analysis was carried out considering a damage criterion based on gas temperature surrounding the electrical equipment higher than 95°C, this criterion was proposed by the licensee;
- 3rd criterion: additionally, still from this reference study, a second sensitivity analysis was performed assuming a damage criterion based on both gas temperature equipment higher than 65°C and soot concentration higher 1.5g/m^3 around the electrical, this criterion coming from the experimental outcomes (see paragraph 3).

In order to explain clearly and simply this approach, an illustrative example is proposed in Figure 8 showing the critical time to achieve the electrical malfunction of an equipment exposed to both gas temperature and soot concentration in its vicinity.

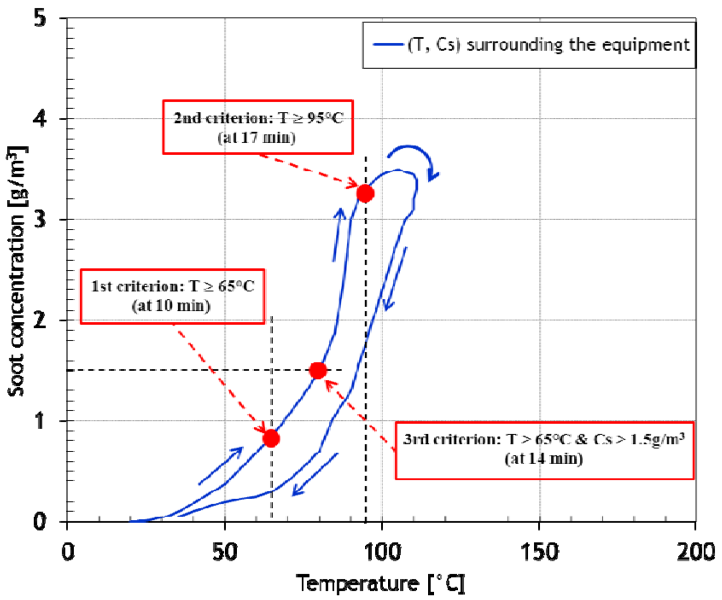


Figure 8. **Illustrative example describing the three types of malfunction criteria considered in fire PSA study for French 1300 MWe French NPPs (blue arrows shows the way of time for temperature and soot concentration)**

From this example, the critical times calculated by SYLVIA code is assessed of 10 min, 14 min and 17 min respectively for the 1st, 2nd and 3rd malfunction criterion. Such numerical outcomes allows the fire PSA study to quantify properly the frequency of core damage due to selected fire scenarios linked with the electrical malfunction criteria of interest. To go further and well understand the fire PSA studies in the French 1300 MWe NPPs based on the work described in this paper, the reader will find the detailed fire PSA study in reference (0).

5. Conclusion

Based on previous knowledge, SYLVIA software tools and experimental tests concerning both electrical cabinet fires and investigations about electrical malfunction of safety equipment, IRSN proposed some original methods to perform calculations in order to assess fire scenarios in the framework of fire PSA studies in the French 1300 MWe NPPs. This study proposes:

- a simple and conservative approach for modelling the fire spread from cabinet to cabinet based on fire tests carried out previously for open and closed-door cabinets (0, 0 and 0);
- three different criteria (in accordance with PSA study) to take into account the electrical malfunction of critical safety equipment due to fire damage. A new approach considers especially the coupling effect of both gas temperature and soot concentration around the equipment, this approach being strongly supported by experimental results performed in real fire scenarios. In order to go further in this promising path, IRSN designed a original furnace that allows investigations on the electrical malfunction of equipment undergoing both gas temperature from ambient to 250°C and soot concentration from 0 to $5 g.m^{-3}$.

From this work, the numerical outcomes of this study were obtained by means of SYLVIA fire code. They have been provided as support studies to assess the fire PSA of the French 1300 MWe NPPs (0). In particular, they also allowed the determination of the most critical safety equipment during fire scenarios in terms of fire-related risk in order to avoid accidental sequences leading potentially to core melting of NPPs.

6. References

[1] Vinot T. (2013), The IRSN approach on fire safety analysis of nuclear facilities, 6th Country Meeting on Fire Safety in Nuclear Installations, Garching (Germany), 30th April 2013

[2] Vinot T., Ormieres Y. and Lacoue J., IRSN global process for conducting a comprehensive fire safety analysis for nuclear installations, 13th International Post-Conference Seminar on "Fire Safety in Nuclear Power Plants and Installations", Columbia (USA), 2013.

[3] Audouin L. et al. (2011), Quantifying differences between computational results and measurements in the case of a large-scale well-confined fire scenario, Nuclear Engineering and Design 241 (18-31), 2011.

[4] Karlsson B. and Quintiere J. (2000), Enclosure Fire Dynamics, CRC Press, 2000.

[5] EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volumes 1 & 2, NUREG/CR-6850, Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, 2005.

[6] Nicoleau F. et al. (2014), Fire PSA and sensitivity studies, proceedings of OECD/NEA International workshop on Fire PRA, 2014.

[7] Coutin M. et al. (2007), Phenomenological description of actual electrical cabinet fires in a free atmosphere, Proceedings of 11th Interflam, 2007.

[8] Rigollet L., Melis S. (2007), HRR of vertical combustibles inside a confinement: An analytical approach to the fire of electrical cabinets, Proceedings of 11th Interflam, 2007.

[9] Plumecocq W., Coutin M., Melis S., Rigollet L. (2011), Characterization of closed-doors electrical cabinet fires in compartments, Fire Safety Journal, 46, pp 243-253, 2011.

[10] Coutin,M., Plumecocq W., Melis S., Audouin L. (2012), Energy balance in a confined fire compartment to assess the heat release rate of an electrical cabinet fire, Fire Safety Journal,52, pp 34-45, 2012.

[11] Bertrand R. et al. (2001), Behaviour of French electrical cables under fire conditions, proceedings of 5th AOSFST, 2001.

[12] Mangs J. (2003), Calorimetric fire experiments on electronic cabinets, Fire Safety Journal 38, 165-186, 2003.

[13] Dreisbach J., Hostikka S., Nowlen S. P., Mc Grattan K. B. (2010), Electrical cable failure – Experiments and simulation, International Conference on Fire Research and Engineering (Interflam), 12, pp 1857-1865, 2010.

[14] Audouin L., Coutin M., Piller M. (2013), Malfunction of electrical equipment due to fire, 6^{th} Country Meeting on Fire Safety in Nuclear Installations, Garching (Germany), 30^{th} April 2013.

[15] Jacobus M.J. (1986), Screening tests of representative NPP components exposed to secondary environment created by fires, NUREG/CR-4596, 1986.

Insights from HRA and MSO analyses in Spanish Fire PRA

Pedro Fernández Ramos
Empresarios Agrupados, Spain

Abstract

Human Reliability Analysis (HRA)

Human reliability analyses in the context of fire scenarios have been substantially improved with the recently published NUREG-1921. This document was the main reference for the updating of HRA analyses for fire PRAs in Spain.

NUREG-1921 presents a systematic approach to the identification and treatment of human action in fire scenarios. It provides methods and criteria to provide screening values as well as to perform detailed analyses and to identify and evaluate dependencies between human actions.

The application of this document requires fully documenting the procedures to be used as information by the operators in support of their decisions to perform the action. It is also important to describe and clearly define the personnel available to perform the action in the main control room or outside the MCR, if required.

An analysis of possible commission errors derived from spurious alarms generated as a result of a fire has also been performed. In this context, only those alarms that require direct action have been considered.

Multiple Spurious Operations (MSO) Analysis

The consideration of spurious actuations is a basic part of the analysis of the consequences of a fire.

In principle, such an analysis should consider any possible spurious actuation that could prevent/affect the performance of a system or a safety function.

The consideration of spurious actuations has been expanded to take into account the possibility of combinations of several spurious actuations that could have specific consequences.

Reference document NEI 00-01 provides a list of possible MSO scenarios that serves as a guide to identify the specific scenarios applicable to a given plant.

The process followed has been to decide which of these scenarios can be considered applicable to a given plant. The analysis was then completed with an expert panel discussion to confirm its applicability and to clarify the expected plant response to each applicable scenario.

The MSO analysis is incorporated into the PRA model by identifying whether the PRA model already includes events that can be considered applicable to the MSO scenario. If this is not the case, the possibility of modifying or including additional events should be considered.

1. Human Reliability Analysis

Operator response in the event of a fire needs to take into account differentiating aspects that are introduced by the fact that the operators need to tackle a potential initiating event caused by the fire (reactor trip, opening of the pressure-relief valves of the pressurizer, etc) while at the same time preparing a response to fight the fire itself. In addition, operators are faced with the problem that certain equipment or indications may malfunction as a result of damage caused by the fire.

Within this context, the operators' response is hampered by the following factors:

- Increased stress
- Lower quantity and quality of the information available in the control room
- Need to spend time planning the fire fighting response
- Depending on the plant, some shift workers may need to leave the control room to manage the fire fighting operations

There may be other additional factors such as the need for additional procedures related to providing a response against a fire.

All these factors have a negative impact on the human actions that are required or listed in the PRA models.

1.2 Human reliability analyses in the event of a fire prior to NUREG-1921

Empresarios Agrupados has performed or collaborated in all the probabilistic fire safety analyses carried out for nuclear power plants in Spain. In the analyses done prior to the publication of NUREG-1921 [1] we strove to take the above aspects into consideration, even if there was no clear methodology for doing so.

In general, the impact of the fire on the probability for human error was considered by identifying the information required to perform the human actions in the procedures in use and by conducting operating personnel interviews.

Every fire scenario identified, among other damage caused by the fire, the part of the above information that could be affected by the fire.

This information was used as a basis to modify the human reliability analyses performed as part of the internal event analyses basically modifying the performance shaping factors depending on the analysis methodology used (HRC or TRC).

1.2 Application of the NUREG-1921 methodology

Human reliability analyses in the context of fire scenarios have undergone substantial improvement with the recently published NUREG-1921 [1]. This document was the main reference for the updating of HRA analyses for fire PRAs in Spain.

NUREG-1921 [1] presents a systematic approach to the identification and treatment of human action in fire scenarios. It provides methods and criteria to provide screening values as well as to perform detailed analyses and identify and evaluate dependencies between human actions.

NUREG-1921 [1] presents a method that is broken down into the following general steps:

- Identification of the operating responses by means of the systematic review of the APS models and the operating procedures

- Confirmation of these responses with the operating personnel to reflect the real plant operation (simulation of scenarios, talk-through)
- Definition of the human errors during the performance of these actions
- Adequate quantification of these human errors following a consistent and objective process
- Analysis of dependencies between human actions
- Adequate documentation of the analysis

These general steps are no different from the steps taken in the human reliability analyses performed prior to the publication of NUREG-1921 [1]. However, this document presents a systematic and structured methodology that ensures the uniformity of the analyses and the results obtained from different plants.

NUREG-1921 [1] applies to the human reliability analyses in the event of a fire. Different methodologies (HCR, TRC) have been applied to the analysis of the human actions included in the internal event PRA analysis and to the calculation of the values of their failure probabilities, so there may be some inconsistencies in the analysis of the same action during a fire and in internal events.

The calculation methods for the values of the cognitive part of human error are the following:

- HCR/ORE for the modelling of non-response based on time available versus time required (only)
- CBDTM for the modelling of non-response based on factors other than timing e.g. availability of cues or procedural issues

The procedure that has been used in the fire PRAs in Spain involves calculating the values with both methods and selecting the highest one.

The experience gained by applying this process indicates that CBDTM yields higher human error probability values than HCR/ORE, except in the cases where the time available for the action is similar to the time required for its execution.

The THERP manual used in the analysis of internal events is applied for the manual part.

The CBDMT method is based on the application of decision trees, which do not allow the values of the selected branches to be weighted. The application of the pc-a ‘Data not available’ tree is especially explanatory:

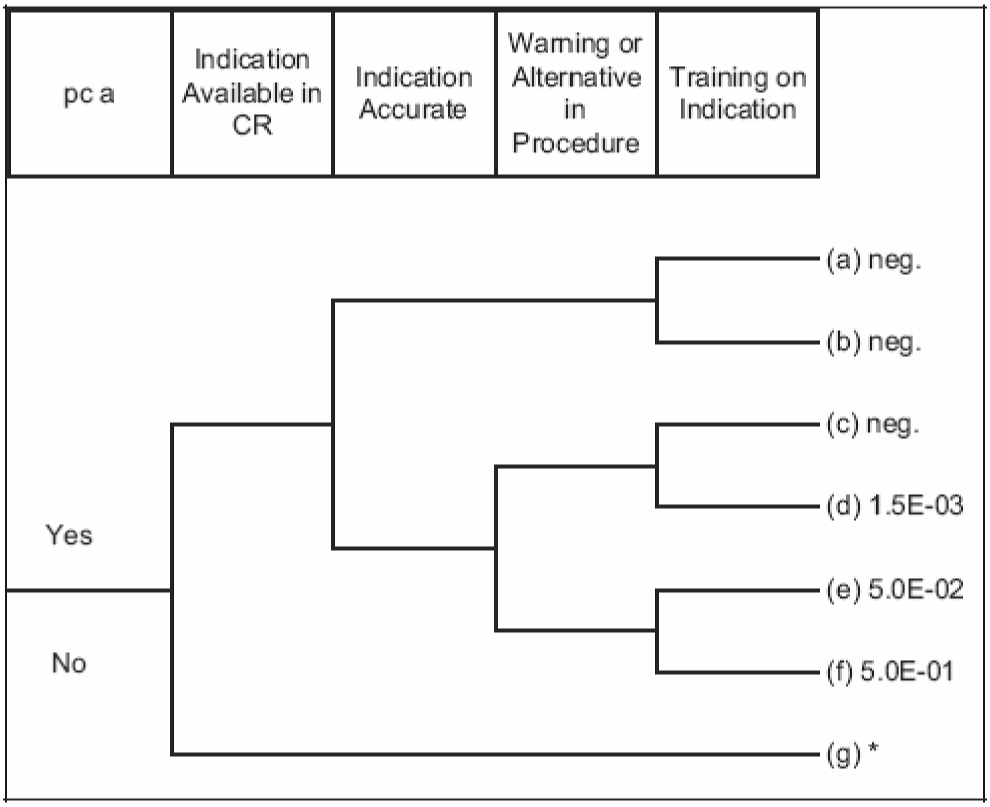


Figure 1. **pc-a "Data not available"**

The description of the shaping factors taken from NUREG-1921 [1] is as follows:

- Are the required indications available in the control room?
  - This is successful if all indications for the specific action are available or if a minimum set of information for the specific action is available.
  - This is unsuccessful if all indications for the specific action are failed. This is the case for total impact: no instrumentation is available, and the HEP evaluates to 1.0.
- Are the indications that are available accurate?
  - The indications are known to be accurate if the fire does not impact any of the instrumentation required for the specific action.
  - The indications are assumed to be inaccurate if there is a partial impact.
- If the normally displayed information is expected to be unreliable, is a warning or a note directing alternative information sources provided in the procedures?
  - The procedure lists alternative instrumentation to perform the specific task or provides a warning of potentially incorrect readings.
  - The procedure provides no alternative instrumentation or a warning. In this case, for existing EOP actions, there are no warnings in the EOPs for fire-related impact on the instrumentation.

- Has the crew received training in interpreting or obtaining the required information under conditions similar to those prevailing in this scenario?
  - The operating crew has received training in interpreting or obtaining the needed information under a fire situation. For cases in which there is partial impact (i.e., a minimum set of instrumentation remains available), the cognitive HEP evaluates to 5.0E-02 if no recoveries are applied.
  - The operating crew has not received training in interpreting or obtaining the needed information under a fire situation. If operators are not trained on performing the EOPs during fire scenarios, the cognitive HEP will evaluate to 0.5 for cases with partial impact on instrumentation if no recoveries are applied.

The decision on the partial assignment of the instrumentation is especially relevant when applying this decision tree. In this sense, it is not possible to indicate the extent of the assigned instrumentation based on the available information, so any damage leads to considering that the instrumentation is not accurate.

Based on this premise, it is important to include indications in the procedures to alert to the possibility of wrong information. However, it is important to note that it is difficult for operating or emergency procedures to alert to this possibility since the damage is specific to each fire area. The availability of aids such as sheets with potentially affected instrumentation and equipment in each area is considered enough to allow operators to make a decision in the event of a conflict in the available indications.

The methodology requires a systematic analysis of all the dependencies between human actions. When dealing with dependencies, the fact of considering the control room as a single location leads to highly conservative results that are probably not very representative of the real dependencies between actions.

1.3 Analysis of responses against spurious alarms

The human reliability analyses that have been performed using NUREG-1921 [1] as a basis include an analysis of potential errors associated to the appearance of spurious alarms caused by fire damage to cables.

The purpose of the analysis has been to check the possibility of the existence of alarms that require a direct response from the operators on the equipment included in the PRA models. These required actuations could change the alignment of required valves or the trip of equipment or switches.

The analysis performed has been based on the information on the alarms in the control room and on the actions required in the event of an alarm.

In some cases, alarms are generated for information purposes. In this case, they are considered to have no effect on the actuation of the operators.

In other cases, alarms are generated for which the operator response would involve additional checks but no direct actuation on any equipment. In this case, it has not been considered that they could lead to additional equipment malfunctions.

The analyses that have been completed up to now have not identified additional equipment malfunctions as a result of spurious alarms in the control room.

2 Analysis of Multiple Spurious Operations (MSO)

A fire-induced spurious operation is an event where the fire causes damage to the cables so that a given component changes to an undesired status that prevents or affects the performance of one or more safety functions.

Since fires can affect several cables, the occurrence of more than one spurious operation during a fire can be contemplated.

The analysis of each fire area identifies all the affected components, either because they failed when performing their active function or because they may suffer a spurious operation.

Combinations of spurious operation in different systems that could affect one or more safety functions have been taken into account.

The concept of multiple spurious operations basically involves the coincidence of several spurious operations, which may occur under different conditions:

- Spurious operations within a same train or redundancy of a system. In this case, the consideration of multiple spurious operations does not present any differences since the end result will be the malfunction of the train or redundancy.
- Spurious operations in different trains or redundancies. In this case, the consideration of more than one spurious operation is the result of considering the malfunction of the affected trains or redundancies.
- Spurious operations in different systems that perform or may perform the same function. In this case, considering more than one spurious operation results in considering a malfunction of alternative systems that perform the same function.
- Spurious operations in different systems or components resulting in a previously-unforeseen scenario. In this case, the result is a different scenario that may be more severe than the already considered scenario(s)

Reference document NEI-00-01 Rev. 2 [2] provides a list of multiple spurious operation (MSO) scenarios that fall within one of the above groups. These lists that have been based on the experience of plants with a similar design are used as a basis for a systematic review of combinations of spurious operations that yield different scenarios to those considered.

Spanish nuclear power plants have been analysed using the above document as a basis to identify any potential scenarios that have not been included in the analyses. Their applicability and potential impact have been analysed.

In some plants, the applicability of the various multiple spurious operation (MSO) scenarios has been addressed in expert panels formed by PRA, electrical, instrumentation and mechanical engineering specialists, as well as by operating personnel.

These analyses have yielded the following results:

- Several scenarios were already included in the PRA models
- Some scenarios have been discarded because of the large number of spurious actuations required or because of specific assumptions or calculations
- Some scenarios have required additional malfunctions that were not initially included in the models to be considered

- Some scenarios have called for certain additional initiating events apart from those initially considered in the models, such as the possibility of having the spurious insulation of the containment isolation valve of a PWR reactor by Westinghouse cause a loss of coolant due to the opening of the safety valve located upstream, in combination with the spurious opening of the automatic isolation valves at the discharge.
- Some scenario resulting from the impossibility of accepting local recovery actions on motor-operated valves.

References

[1] Electric Power Research Institute (EPRI) and Nuclear regulatory Commission Office of Nuclear Research (NRC-RES), (2012m) EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, NUREG-1921, EPRI 1023001, Washington, D.C., Palo Alto, CA, USA

[2] Nuclear Energy Institute (NEI), (2009), Guidance for Post-Fire Safety Shutdown Circuit Analysis, Final Report, NEI 00-01, Revision 2

Fire PSA and Sensitivity Studies

F. Nicoleau, F. Corenwinder, G. Georgescu
Institute for Radiological Protection and Nuclear Safety (IRSN), Fontenay-aux-Roses, France

Abstract

IRSN (TSO of French Nuclear Safety Authority) develops level 1 probabilistic safety assessments (PSAs) for nuclear power plants (NPPs) in order to establish his own independent opinion on the assumptions and results of the licensee (EDF) PSAs. In this context, IRSN has developed a Fire Level 1 probabilistic safety assessment (PSA) for 1,300 MWe reactors. The study is an extension of the IRSN in-house 1,300 MWe nuclear power plants (NPPs) Level 1 PSA for internal events. The IRSN objectives are to provide an independent verification of the EDF study and also to allow further PSA applications in the framework of technical instruction of safety issues.

The IRSN Fire PSA is a focused scope study, since only the most important fire compartments are included. This study takes into account the failure of equipment and cables due to the fire and the impact of fire on operator actions. IRSN also performed some investigations in the R&D area, like researches on damage temperature criteria and on the impact of the smoke on component failures. The fire simulation, with the IRSN code SYLVIA (a two-zones fire model code) is used to determine the evolution of the temperature in the compartments, including the adjacent compartments in case of fire spreading, in order to estimate when the components may fail (failure time). The IRSN PSA model is also used as a tool to perform sensitivity studies, presented in this paper, in order to estimate the impact of the choice of the component damage criteria or to highlight potential impact of the electrical faults propagations.

1. Introduction

The periodic safety review procedure is a periodic process implemented for every reactor. In France, the periodic safety reviews occur every ten years and concern all reactors of a given serie (e.g. 900 MWe or 1,300 MWe or 1450 MWe reactors). The use of PSA for periodic safety review is done accordingly with the French PSA Basic Safety Rule [1].

In the context of the third decennial visit for the French 1,300 MWe nuclear power plants, IRSN has developed a fire level 1 PSA for 1,300 MWe reactors. IRSN objectives were to provide his own model in order to evaluate assumptions and results of the licensee Fire PSA. The study is an extension of IRSN in-house 1,300 MWe NPPs Level 1 PSA, for internal events. The development of a Fire PSA is necessary due to the importance of fire on the risk of core damage.

A lot of information was exchanged between the licensee and IRSN during the development of the project. The licensee and IRSN studies are similar in scope and use the same principles; however the objectives and the main assumptions may be different (for example: damage temperature considered for the equipment, fire source characteristics…). In particular, IRSN objective is to identify and to quantify preponderant accident sequences leading to core melt. The study will therefore focus on the most critical equipment and compartments in terms of fire-related risks.

IRSN objectives are also to provide an independent verification of the licensee study and to use Fire PSA applications in the framework of technical instruction on specific safety issues.

The IRSN PSA model is also used as a tool to perform sensitivity studies, presented in this article, to estimate the impact on the results of the choice of the component damage criteria or to highlight the risk impact of the electrical faults propagations.

2. Specifics of the French Context

Regarding nuclear industry, France represents a unique situation with a rather large fleet of Nuclear Power Plants (58 in operating, 1 in construction) which are all built by the same manufacturer (AREVA) and operated by the same licensee (EDF). This nuclear fleet is standardized in 3 PWR series - soon 4 with EPR – (900 MWe: 34 plants, 1,300 MWe included two types plants named P4 and P'4: 20 plants, 1,450 MWe: 4 plants; EPR: 1 plant). The PWR plants of a given series are almost identical in design and operation. The standardized series has real advantages in terms of experience feedback. In the specific field of PSA, the situation is particularly favorable for data collection, and moreover a single PSA (at least for level 1 PSA and internal initiating events) is sufficient for whole PWR series of plants. Since few years, IRSN has begun to develop also PSA for internal hazards in order to identify potential improvement in design and operation of NPP and to assess similar studies developed by the licensee. Regarding PSA for external events, developments are still ongoing especially for seismic hazards and other external hazards inducing long term loss of offsite power and heat sink.

Concerning IRSN Fire PSA, two models have been developed. The first development of Fire PSA started in the 90's and concerned French 900 MWe. This study was achieved in 2007. It was a very complete study, developed as recommended in the international practice [2]. It will be updated in 2014 to take into account new data and new experience feedbacks. Moreover, the model will be completely reviewed and implemented with Risk Spectrum tool® in order to facilitate sensitivity studies.

The second development of Fire PSA started in 2005 and concerned French 1,300 MWe reactors. The general method adopted by IRSN for its 1,300 MWe Fire PSA is similar to the one used for 900 MWe reactors Fire PSA. Nevertheless, the lessons learned from the development of the 900 MWe reactors Fire PSA, as well as the progress in computer tools, have led to some improvements which are presented in this paper.

3. General Method

The IRSN fire PSA methodology, developed for 1,300 MWe reactors, based on international practice [2], is divided into three steps (e.g. figure 1). The step details are provided in the article [3].

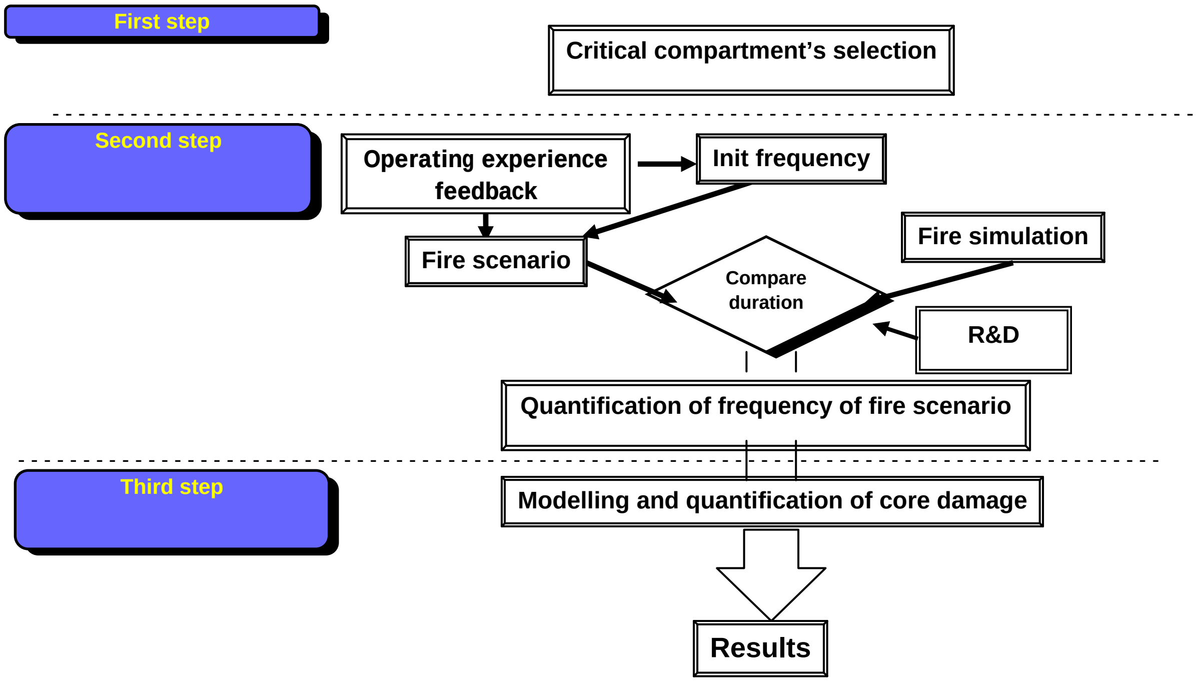


Figure 1. **General approach for conducting the IRSN Fire PSA**

3.1 First step: critical compartments selection

The fire compartments screening process is based on the evaluation of "*importance calculations*" performed using the internal events level 1 PSA for 1,300 MWe reactors and on experience feedback from the development of IRSN 900 MWe Fire PSA; this Fire PSA showed that the critical compartments are located mainly in the electrical building.

The approach used to develop the 900 MWe Fire PSA was very exhaustive and each compartment of the NPP was studied in details. Finally, most of the selected compartments for detailed analysis were located in the electrical building.

For the 1,300 MWe plants Fire PSA, due to limited resources and short time available to perform the study, another method was chosen. This method is based on the identification of the critical targets. A target is a safety related component whose failure leads to an initiating event or to a mitigation system failure. To identify the critical target contributing to the core damage, importance calculations with the RiskSpectrum® PSA software for the level 1 "internal events" PSA were done.

These calculations pointed out the components which are dominant for the internal events level 1 PSA, what allows to identify the fire compartments, potentially critical, containing these components. A critical compartment is a compartment containing targets or which is adjacent to a compartment containing targets.

3.2 Second step: quantification of fire scenario frequency

The second step is the modelling and the quantification of fire scenario. Many data are needed for the modelling, mainly for fire scenarios and fire simulations. Data were collected mainly by the licensee for the studied NPP (the study was done for one NPP in particular). These data were completed by IRSN, for some specific technical information after a plant walkdown (e.g. characteristics of the ventilation system like pressure, loss coefficients and compartment geometry like size, openings and their location in the room…).

The collected operating experience related to fire incidents covers the period from April 21, 1975 to December 31, 2009. It represents more than 830 fire events for 1401 years reactor. For the 1,300 MWe Fire PSA, 719 departures of fires were selected (self-extinction is not taken into account) in order to estimate statistical parameters like fire frequency of equipment, failure rate of fire protection systems (e.g. fire dampers), time needed for the intervention teams to extinguish fire and human factor errors probabilities (e.g. operating, intervention, etc. …). Those statistical parameters are used to quantify the fire frequency of critical compartments and the failure probability of human action for detection or extinction.

For each fire scenario, the consequence on the installation is determined. It's then possible to gather all fire scenarios with the same consequence on the plant and to estimate their frequency. The list of the components and the electrical cables lost, in the fire compartment and in the adjacent compartments due to the fire spreading, is putted in the consequence. To establish this list, fire simulation is necessary. IRSN develops a two-zone fire model, called SYLVIA code. It's a software system simulating fire, ventilation and aerosol contamination phenomena. The SYLVIA code allows to estimate the failure time (time when the components fail) of various components in the fire compartment and in the adjacent compartments in case of fire spreading.

To estimate if a component is lost during the fire scenario, a comparison is done between the failure time and the duration of fire. If the failure time is lower than the fire duration, the component is lost. In the other case, the component is available. This method gives the list of components or cables which are lost.

3.3 Third step: quantification of frequency of core damage

The objective of the third step is to model and quantify core damage sequences induced by a fire. The model is based on the existing "internal events" level 1 PSA. All event trees are adapted in order to incorporate the fire fighting features and accident procedures, for example the fire accident procedures which are implemented for each fire compartment of French NPP.

The Fire PSA model is entirely developed using RiskSpectrum® PSA software, thus giving overall coherence to the model, facilitating sensitivity studies on the most important assumptions.

The model is composed by three types of events trees. Firstly, an event tree is used for the modelling of the fire scenario. Then, an intermediate event tree between the fire scenario event tree and an adapted event tree of 1,300 MWe "internal events" level 1 PSA is developed to take fire scenario specificities into account. The figure 2 presents an example of links between the event trees.

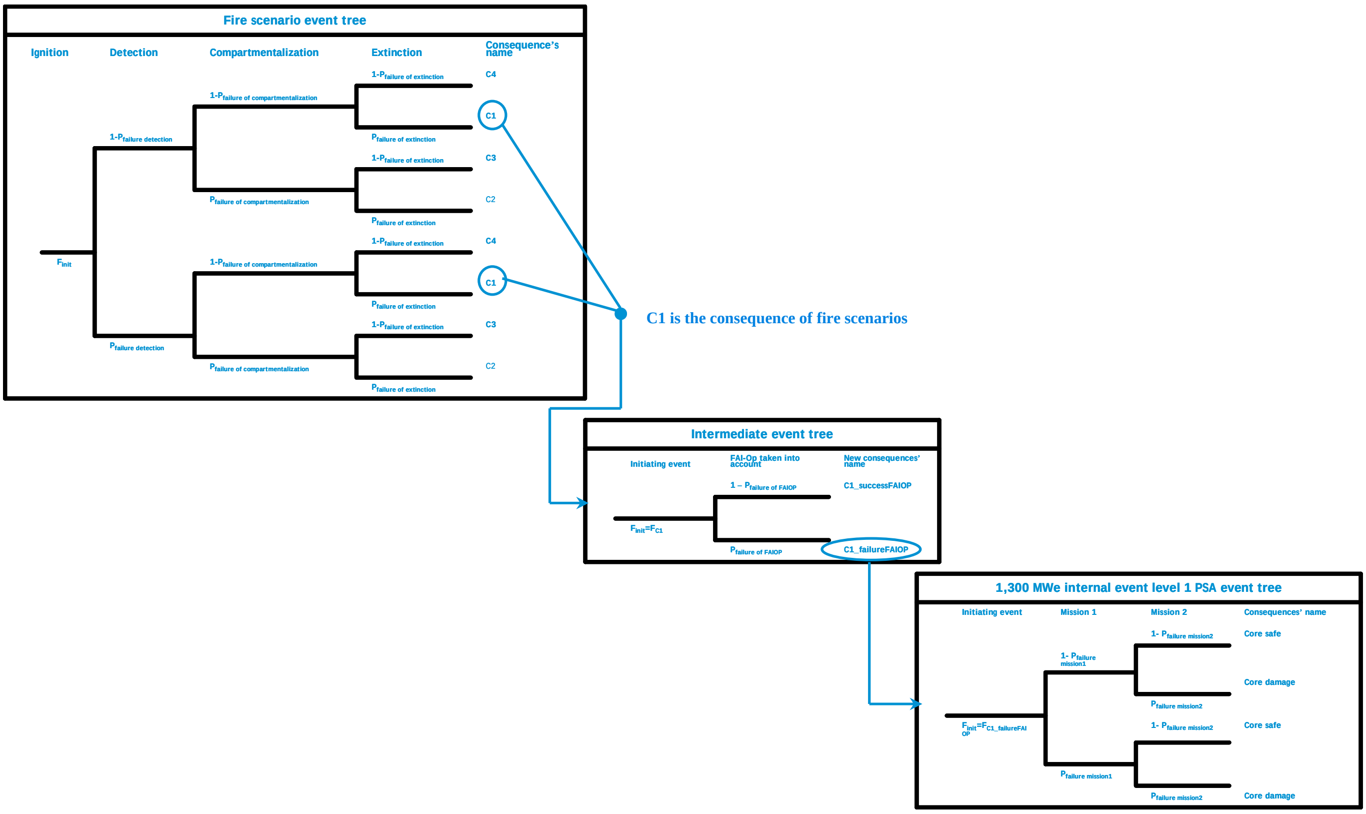


Figure 2. **Architecture of the model: link between the event trees of Fire PSA**

The intermediate event tree is a specificity of the IRSN Fire PSA. It makes it possible to take into account the success or the failure of operator actions and of the fire fighting features in case of fire. The intermediate event tree are developed taking into account the specific fire accident procedures for the concerned fire compartment, for example, the actions to be realized by the operator in the main control room (MCR). The fire accident procedure is applicable when the fire is confirmed by the chief of the on-site plant fire team. It's divided into several parts: 'measure', 'anticipated action' and 'total power supply cut-off' including the batteries cut-off in order to avoid spurious alarm caused by fire.

The link between the event trees is made by using the consequence of each sequence of fire scenario. The fire scenario's consequence, for example the consequence called "C1" in the figure 2, indicates the frequency of all fire scenarios with the same consequence and the list of components and electrical cables lost in the critical compartment and in the adjacent compartment. RiskSpectrum® PSA software calculates the frequency for each fire scenario as well as for the set of fire scenario that have been assigned the same consequence. The result of the fire scenario event tree, for example the consequence "C1", is the input of the intermediate event tree which takes into account the success or the failure of the human action such as actions to avoid spurious signals in the Main Control Room. The consequence of the intermediate event tree, called "C1_failureFAIOp", is the input of the 1,300 MWe "internal events" level 1 PSA event tree. In RiskSpectrum® tool, the consequence (the list of the lost components) is then assigned as a "boundary conditions set" (BCSET) for the corresponding internal events PSA initiating event.

The approach by consequence allows to perform easily sensitivity analysis on data or assumptions and to obtain the different contributions of the risk. As an example, two performed sensitivity studies are presented in the next paragraph.

4. Sensitivity Studies

IRSN performed several sensitivity studies, mainly in order to improve the modelling, modifying some "uncertain" values and estimating the effect on the core damage frequency and on the contributions of each critical fire compartment. The sensitivity studies allowed also to determine the effect of uncertainties on the overall fire risk. For example, the sensitivity studies allowed to estimate the impact on RFC of the fire development modelling (fire growth, conduction, heat release rate …), of the damage temperature of components, of the failure of human factor for doing the actions of the fire accident procedure …

Two sensitivity studies are presented below:

- The first sensitivity study concerns the choice of damage criteria of electrical and I&C cabinet,
- The second sensitivity study concerns the electrical faults propagation, potentially leading to the loss of the auxiliary transformer and of the "service" transformer, in case of fire in non-classified switchboard in the electrical building.

4.1 First sensitivity study: damage criteria of electrical and I&C cabinet

The first study concerns the choice of the damage criteria of electrical and I&C cabinets. It has to be noted that in general this parameter is highly uncertain. In international R&D [2], values are proposed but due to the lack of knowledge on the failure of electronic and electric cards contained in electrical and I&C cabinet, the behaviour in case of fire remains uncertain. IRSN is performing R&D program about the damage temperature of electrical cabinet [4] but the tests are still going on. The first analysis of the latest tests showed that the electronic cards were lost at a temperature value (65°C) lower than the value of temperature found in the first experimental program done in 2009 (higher than 100°C) because in the latest tests, the ambiance is more representative than a real fire with the presence of soot (a real electrical cabinet was in fire). Another conclusion was that the electronic cards fail when some conditions on temperature and soot are reached: a combination of values of two "damage criteria" (temperature and soot) could cause the electronic cards' malfunction.

Taken into account the conclusion of the IRSN R&D program and the value retained by the licensee for the damage criteria (criterion only in terms on temperature: upper or equal to 95°C, soot impact not considered), a sensitivity study is performed with different damage criteria for electrical and I&C cabinet:

- For the reference study, the damage criterion is the achievement of a temperature (named T) of 65°C; it is assumed to be conservative,
- For the first sensitivity test, the damage criterion is the achievement of a temperature (named T) of 95°C (the same value than the one considered by the licensee),
- For the second sensitivity test, the damage criterion is the achievement of a temperature (named T) of 65°C **and** a concentration of soot, name Csoot equal to 1,5g/m^3.

Five critical compartments are studied, called ROOM1, ROOM2, ROOM3, ROOM4 and ROOM8. They correspond to the compartments having the most important contribution in core damage frequency (CDF) for the reference study. The five critical compartments and their adjacent compartments are shown in the figure 3.

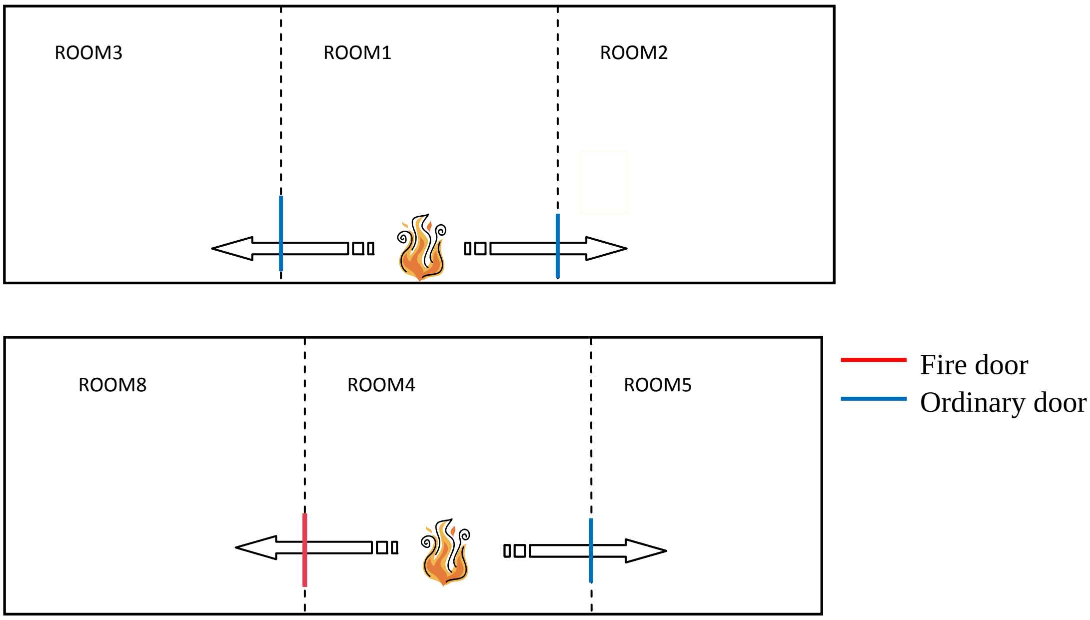


Figure 3. **Presentation of compartments studied**

Several simulations were realized with SYLVIA tool to estimate the temperature and the concentration of soot in the compartment in fire and in the adjacent compartments. One simulation is done for the compartment in fire and for one adjacent compartment with the doors in communication considered opened.

The next table shows the main consequences of a fire in each compartment for each test and the evolution of CDF.

Table 1. **Results for the first sensitivity study**

Compart-ment in fire	**Reference study Damage criterion: T=65 °C**	**Test 1 Damage criterion: T = 95 °C**		**Test 2 Damage criterion: T = 65 °C and Csoot = 1,5g/m^3**	
	Consequence of fire	**Consequence of fire**	**CDF reduced (%)**	**Consequence of fire**	**CDF reduced (%)**
ROOM1	Temperature ≥ 65 °C at 15 minutes for the room in fire and in the adjacent rooms 2 initiating events : • loss of one train • loss of external electricity	Temperature in the adjacent rooms < 95 °C Temperature ≥ 95 °C at 17 minutes for the room in fire No initiating event	100	Same consequences than the reference study (**to be conservative**) with 5 minutes of difference for the failure of components in the adjacent rooms	-

Compart-ment in fire	**Reference study Damage criterion: T=65 °C**	**Test 1 Damage criterion: T = 95 °C**		**Test 2 Damage criterion: T = 65 °C and Csoot = 1,5g/m^3**	
	Consequence of fire	**Consequence of fire**	**CDF reduced (%)**	**Consequence of fire**	**CDF reduced (%)**
ROOM2	Temperature ≥ 65° C at 12 minutes for the room in fire and the adjacent room 1 initiating event : • loss of one train	Temperature ≥ 95 °C at 13 minutes for the room in fire, not in the adjacent room, temperature is still under 9 5°C 1 initiating event : • loss of one train	79	Temperature in the adjacent room ≥ 65 °C and Csoot ≥ 1,5g/m^3 at 17 minutes for the room in fire and 22 minutes for the adjacent room 1 initiating event : • loss of one train	82
ROOM3	Temperature ≥ 65 °C at 15 minutes for the room in fire and in the adjacent room 1 initiating event : • loss of external electricity	Temperature < 95°C in the adjacent room Temperature ≥ 95°C at 17 minutes for the room in fire No initiating event	100	Same consequences than the reference study (**to be conservative**) with 5 minutes of difference for the failure of components in the adjacent room	-
ROOM4	Temperature ≥ 65 °C at 15 minutes in the room in fire and the adjacent rooms 2 initiating events: • loss of 380 V cabinet (in ROOM5) • loss of 380 V cabinet (in ROOM8)	Temperature < 95 °C in the adjacent rooms Temperature ≥ 95°C at 16 minutes for the room in fire No initiating event	100	Same consequences than the reference study (**to be conservative**) with 5 minutes of difference for the failure of components in the adjacent rooms	-
ROOM8	Temperature ≥ 65° C at 15 minutes for the room in fire and at 22 minutes in the adjacent room 1 initiating event: • loss of 380 V cabinet (in ROOM8)	Temperature < 95 °C in the adjacent room Temperature ≥ 95 °C at 15 minutes for the room in fire 1 initiating event: • loss of 380 V cabinet (in ROOM8)	95	Temperature≥65°C and Csoot ≥1,5 g/m^3 at 15 minutes for the room in fire and at 30 minutes in the adjacent room 1 initiating event: • loss of 380 V cabinet (in ROOM8)	91
Total	-	-	92	-	48

Nota for a fire in the ROOM2: the CDF of test 1 is upper than the CDF of test 2, in comparison with the reference study, because the adjacent room ROOM1 doesn't contain safety related equipment (there is no impact on CDF, if the equipment of ROOM1 are lost or not lost) and the failure time of all equipment in the ROOM2, is bigger in test 2.

The global CDF decreases in total by 92 %, if the damage temperature is increased by 30 °C and decreases in total by 48 %, if the damage criterion takes into account temperature and soot.

If the damage temperature is increased by 30 °C, the component are lost later or fewer components are lost in the fire compartment and in the adjacent compartment and less initiating events are induced and less safety related equipment and mitigation equipment are lost.

This sensitivity study shows that the impact on CDF of the value of damage temperature seems more important than the impact of a combination of temperature and soot criterion to evaluate CDF.

4.2 Second sensitivity study

The objective of this sensitivity study is to estimate the impact of the design features to avoid electrical faults propagation, in case of cable fault, following a fire on non-classified switchboard in electrical building. In case of a fire damaging a non-classified switchboard or the cables connecting to non-classified switchboard with the service transformer (called "TS") or with the auxiliary transformer (called "TA"), the generated short circuit can involve, due to the electric protections, the opening of the circuit breakers which supplies the non-classified switchboard from the "TS" or the "TA" and can potentially lead to the loss of the auxiliary transformer and of the service transformer.

The next figure shows the electrical cables between the switchboards and the transformers:

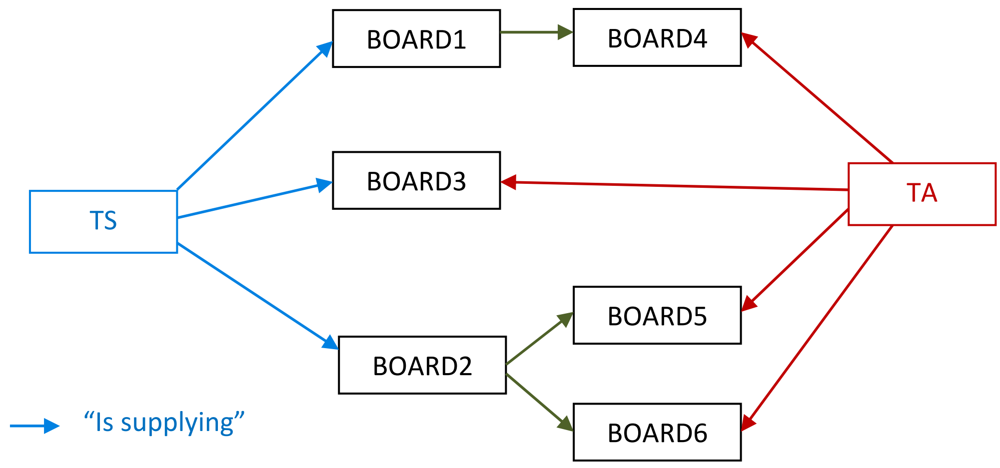


Figure 4. **Presentation of electrical cables between switchboards and transformers**

The sensitivity study takes the following hypothesis: the "TS" and/or the "TA" are supposed to be lost if a fire occurs on a switchboard supplied by the transformer(s). For example, the switchboard, called BOARD3, is supplied by the "TS" and the "TA". The "TS" and the "TA" are lost if a fire occurs on this switchboard.

Only compartments, containing cables supplying switchboards from the "TS" or the "TA" are considered for this study. These are the same compartments retained in the first sensitivity study, called ROOM1 and ROOM3 (see Figure 3). The table 2 presents the consequences of the fire in each compartment taking the loss of the transformers into account and the increased of CDF compared with the reference study.

Table 2. **Results for the second sensitivity study**

Fire scenario	**Sensitivity study**		**Reference study**	**Increased of CDF**
Fire scenario [Compartment in fire-number of scenario]	**Component failure due to fire**	**Component available**	**Difference between two studies**	**compared with reference study**
ROOM3-C4	Loss of trains A and B	Only the diesel train A and B and LLS	BOARD3 and BOARD6 are not lost. They supply power cabinet of trains A and B. Trains A and B are Ok	987 %
ROOM1-C4	Loss of trains A and B	Only the diesel train A and B and LLS	BOARD3 and BOARD6 are not lost. They supply power cabinet of trains A and B. Trains A and B are Ok.	717 %
ROOM1-C6	Loss of trains A and B	Only the diesel B and LLS	BOARD6 is not lost. Loss of train A	442 %

The increase of the CDF, due to the cable defect in case of fire on switchboard, shows that the loss of both transformers is very penalizing for the two compartments. Nevertheless, the global CDF (considering all compartments) increases only by a factor two if the cable fault is taken into account, because only two compartments are concerned for this sensitivity study.

5. Conclusion

The periodic safety review procedure is a periodic process implemented for a given reactor series. PSAs are an important aspect of the periodic safety review for the French 1,300 MWe nuclear power plants which is on-going. In this context, IRSN develops, on a specific plant, a Fire PSA for 1,300 MWe NPPs in order to establish its own independent opinion on the assumptions and the results of the Fire PSA that will be conducted by the licensee.

The objective of this article was to present the methodology developed for the IRSN Fire PSA and the sensitivity studies performed.

The sensitivity studies highlight the importance of the choice of damage criteria and the relevance of protective measures to avoid cable fault in case of fire on non-classified switchboard in electrical building that would involve the loss of the auxiliary or of the service transformer.

To conclude, it is important to note that during the first steps of development of a Fire PSA, hypothesis more or less conservatives, as well as parameter value with various uncertainties are used or component failure or cable fault. It’s very important to analyse the effects of those choices on the PSA results and to identify the possible cliff-edge effects and the needs on R&D led.

References

[1] ASN (2002), “French PSA Basic Safety Rule 2002-01”

[2] EPRI/NRC-RES, “Fire PRA Methodology for nuclear Power Facilities”, NUREG/CR-6850

[3] F. Nicoleau (2010), “Fire PSA for French 1,300 MWe NPPs”, article PSAM10

[4] J. Espargillière, T. Vinot, L. Audouin (2014), “Focus on the studies in support of fire-PSA of the French 1,300 MWe Nuclear Power Plants”, article for the International Workshop on Fire PSA in Garching, Germany

Fire PRA and Multiple Spurious Operations Study in Korea UCN 3 unit

Kilyoo Kim, DaeIl Kang, Seung-Cheol Jang
Integrated Safety Assessment Division, Korea Atomic Energy Research Institute
P.O. Box 105, Yuseong, Daejeon 305-600, South Korea
kykim@kaeri.re.kr

1. Introduction

A fire PRA adapting NUREG/CR-6850 [1] methodology is being done for UCN 3 unit in Korea. The possible fire-induced multiple spurious operations (MSOs) are reviewed as recommended in NEI-00-01 [2] and possible combination sets of equipment failures causing MSOs are derived by fault trees. By using the combination sets of equipment failures and by checking all electric cables of each fire compartment, possible MSOs by the fire compartments(~rooms) are derived. In finding MSOs by the fire compartments, the type of cables such as motive power, control power, signal is considered, and the pumps and valves are differently treated.

The fire PRA model is automatically developed based on the internal PRA model by using the software called IPRO-ZONE [3], which can avoid a tedious manual fault tree modification. An automatically modification from internal PRA to Fire PRA including MSOs using IRPO-ZONE is in detailed described in this paper.

The quantification of fire PRA model is performed by 'one-shot' quantification [4] which is faster quantification. Therefore, the fault trees are automatically modified for the 'one-shot' quantification to be easily performed.

2. Description of the Actual Work

2.1 Procedure of reflecting MSO in fire PRA

The following steps are the procedure of reflecting MSO in a fire PRA.

1) Review of the generic MSO scenarios suggested in NEI-00-01.
2) The MSO equipment combinations are derived by modelling fault trees(FTs)
3) MSO rooms are derived where fire occurrence could cause MSO
4) Prepare a mapping table which shows that each MSO room fire is related with initiating events and equipment failures.
5) A fire PRA model is automatically developed with the mapping table by IPRO-zone software.

Each step is further explained as follows.

2.2.1 Review of the generic MSO scenarios

After reviewing the 56 generic MSO scenarios of NEI-00-01, it is determined that 22 MSO scenarios could be applicable to UCN 3. Loss of RCP seal cooling(MSO-1), MSO of reactor vessel head vent valves(MSO-20) and MSO of atmospheric dump valves(MSO-23), etc. are typical MSO examples in UCN 3.

2.2.2 MSO equipment combinations

For each MSO scenarios, MSO equipment combinations are found by developing and quantifying a FT. 193 components are related to the combinations, and 1,907 cables are related to the components. An example FT is shown in Fig. 1. In Fig. 1, a FT to find equipment combinations which cause MSO-20 scenario is modeled. The founded equipment combinations are described in Table 1.

2.2.3 MSO rooms

After finding the MSO equipment combinations, the next step is to find MSO rooms in which fire occurrence causes the failure of all MSO equipment combinations. The MSO rooms are found by the cable information such as cable paths, connected equipment, etc. An illustrative cable data is shown in Table 2. In Table 2, six cables are passing and one pump (9441MPP02A) locates in room 047-A03A.

An example of selected MSO room for MSO-20 scenario is shown in Table 3. In Table 3, room 144-A01 is one of MSO room for MSO-20 scenario, and 144-A03A is not one since no cable for valve 9433V0101 is passing in room 144-A03A. In determining the MSO rooms, some cables should be neglected whose fire cannot make spurious operations. That is, the cables connected to space heaters, annunciation for lube oil level, etc. could be neglected.

40 rooms among 179 rooms are selected for the possible MSO rooms which have 458 equipment combinations for 17 MSO scenarios.

2.2.4 Mapping table of MSO Fire rooms

In a Fire PRA, one necessary procedure is to set up a mapping table which shows the affected components and their failure modes, etc., when a fire occurred in each room. With this mapping table, we can automatically generate a fire PRA model from an internal PRA model by using IPRO-Zone since IPRO-ZONE automatically modifies the FTs of the internal PSA by reflecting the information of the mapping table.

Illustrative 3 mapping tables adjusted to IPRO-ZONE, are shown in Table 4. IPRO-ZONE reads the 3 tables shown in Table 4, and generates SIMA files to be used in the AIMS of PSA software. For example, the 2nd row of iZone table in Table 4, is interpreted to the SIMA script as shown in Table 5, and later AIMS-PSA reads the SIMA script and converts the FT of internal PRA as shown in Fig. 2 to the FTs of fire PRA as shown in Fig. 3.

Actually, MSO-23 scenario could occur in 165-A01A room since there are the cables for ADV 171, 172, and 173 in the room. However, since the atmosphere dump valve (ADV) for UCN 3[5] has 3 parts as mentioned in Table 6, all parts should be spuriously open before the ADV fails closed(fail-safe). With the data given in Ref[6], for 3 minutes duration, the conservative conditional probability of LSSBOUT (= Large Secondary Side Break, OUTside of containment) induced by the fire of 165-A01A room is 6.96-3, and thus, the LSSBOUT induced by MSO in 165-A01A room is conservatively 3.19E-6(=6.96E-3 * 4.59E-4) by multiplying the ignition frequency of 165-A01A room. In Table 6, ADV 173 was neglected since their tray was well insulated.

If a MSO in the same room causes different initiating events, then different zone names are used. For example, 165-A01A is used for %ILOFW, and 165-A01ALA for %ILSSBOUT as shown in iZone table of Table 4.

3. Results

A systematic derivation of MSOs in each room was performed, and its procedure is described. The MSO rooms can be easily included in the fire PRA by using IPRO-ZONE after preparing a mapping table.

Acknowledgement

This work was supported by Nuclear research & Development Program of the National Research Foundation of Korea (NRF) grant, funded by the Korean government, Ministry of Science, Ict & future Planning (MSIP).

References

[1] U.S. NRC, NUREG/CR-6850 (EPRI 1011989), Fire PRA Methodology for Nuclear Power Facilities, September 2005

[2] NEI-00-01, “Guidance for Post-Fire Safe-Shutdown Circuit Analysis,” Revision 3(draft), Nuclear Energy Institute, May 2010

[3] Dail Kang, et al., "Development of the IPRO-ZONE for internal fire probabilistic safety assessment", Nuclear Engineering and Design 257 72-78, 2013

[4] Dail Kang, et al., “An approach to the construction of a one top fire event PSA model”, Nuclear Engineering and Design 239 2514-2520, 2009

[5] ENERTECH, “Installation, operation, maintenance manual for Electro-Hydraulic Actuator (for Atmosphere Dump Valves) for Ulchin Nuclear Power Plant”, MA 21901, June 2011.

[6] NRC, “Supplemental interim technical guidance on fire induced circuit failure mode likelihood analysis”, Feb. 2014

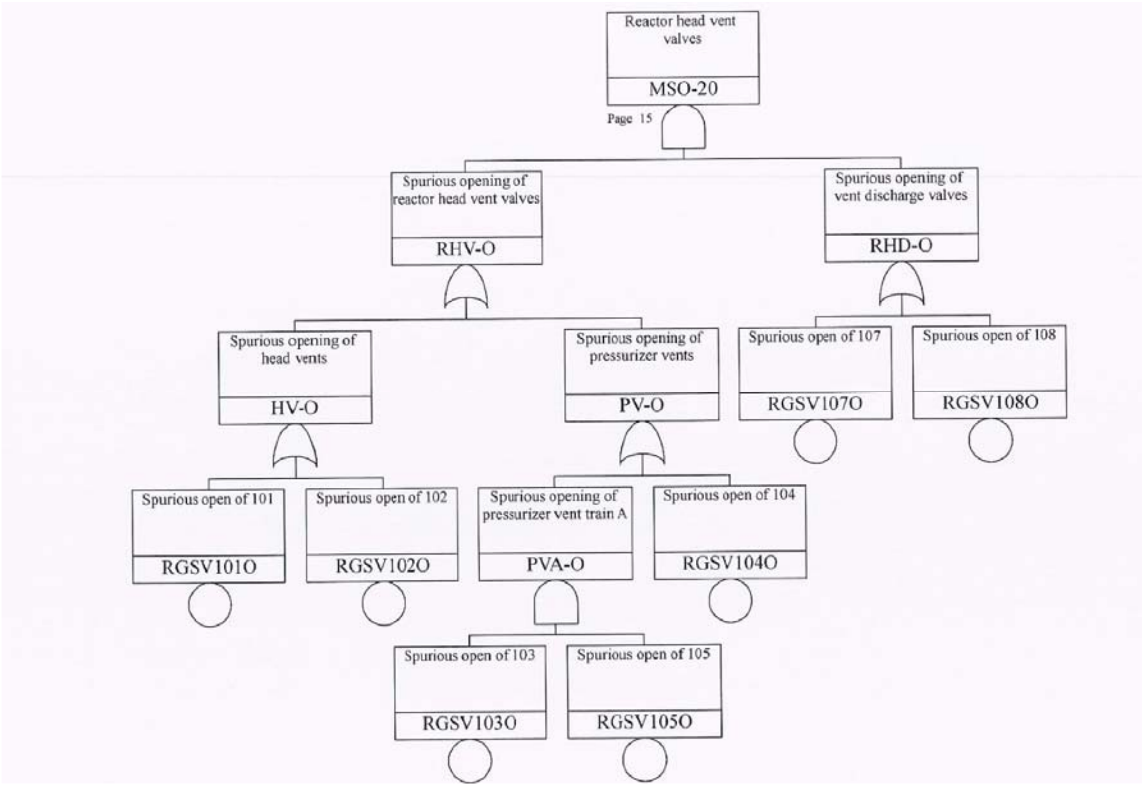


Figure 1. **An example fault tree to find equipment combinations for MSO-20 scenario**

Table 1. **MSO Equipment combinations for MSO-20 scenario**

MSO scenario	MCS	Equip 1	Equip 2	Equip 3
MSO-20	1	RGSV101O	RGSV107O	
	2	RGSV101O	RGSV108O	
	3	RGSV102O	RGSV107O	
	4	RGSV102O	RGSV108O	
	5	RGSV103O	RGSV105O	RGSV107O
	6	RGSV103O	RGSV105O	RGSV108O
	7	RGSV104O	RGSV107O	
	8	RGSV104O	RGSV108O	

Table 2. **An example of cable information**

(green)	: Equip. locates inside Room

Room NO : 047-A02A

Cable NO	EQ	EQNAME	STARTEQ	ENDEQ	DIV	CABLETYPE
3442C01ABAA	9442MPP01A	CONTAINMENT SPRAY PP A	9442MPP01A	9823ESW01A	A	P

Room NO : 047-A03A

Cable NO	EQ	EQNAME	STARTEQ	ENDEQ	DIV	CABLETYPE
3441C01CBAA	9441MPP02A	HPSI PP	9441MPP02A	9823ESW01A	A	P
3442C01ABAA	9442MPP01A	CONTAINMENT SPRAY PP A	9442MPP01A	9823ESW01A	A	P
3451C01AEAA	9451MPP01	CHARGING PP	9451MPP01	9825ELC01A	A	P
3451C01AJCA	9451MPP01	CHARGING PP	9752JPA03A-02R	9451W001A	A	C
3451C01CEAA	9451MPP03	CHARGING PP	9451MPP03	9825ELC02A	A	P
3451C01CJCA	9451MPP03	CHARGING PP	9752JPA03A-02R	9451W002A	A	C

Table 3. **An example of MSO room for MSO-20 scenario.**

MSO-20			
	MCS : 1	RGSV101O	RGSV107O
	P&ID Name →	9433V0101	9433V0107
Room NO :	144-A01	O	O
	Cable NO	CS1	CS2
	3433C01ASAA	O	
	3433C01FSAA		O
…	…	…	…
Room NO :	144-A03A	X	O
	Cable NO	CS1	CS2
	3433C01FJCA		O

Table 4. **An example of mapping tables used in IPRO-Zone**

iZoneEquip

Zone	EquipID	ZoneEquipType
165-A01ALA	9521V0171	PT^MCT
165-A01ALA	9521V0172	PK^MCK
…	…	…

iZone

Zone	Path	TransferZone	Frequency	EventTree	BarrierProba	SeverityProba	NonSupProba
165-A01A			0.000459	%ILOFW		0.2	
165-A01ALA			3.19e-6	%ILSSBOUT		0.2	
165-A01ALA	CAB	182-PAB	3.19e-6	%ILSSBOUT	0.0012	0.2	0.51
165-A01A	CAB	182-PAB	0.000459	%ILOFW	0.0012	0.2	
…	…	…	…	…	…	…	…

iEquipEvent

EquipID	Equip Code	PSAEvent	PSA Event Description	Normal Position	Desired Position	Failed Position
9521V0171	PV	MSEVCADV171	ADV 171 FAILS TO CLOSE	OPEN	CLOSE	FC
9521V0171	PV	MSEVO171	ADV 171 FAILS TO OPEN	CLOSE	OPEN	FC
9521V0172	PV	MSEVCADV172	ADV 172 FAILS TO CLOSE	OPEN	CLOSE	FC
9521V0172	PV	MSEVO172	ADV 172 FAILS TO OPEN	CLOSE	OPEN	FC

Table 5. **SIMA Script generated by IPRO-Zone**

```
;[Initialing Event]
;SeverityProba=0.2, NonSupProba=1
Add+  %ILSSBOUT       G&F-%F-165-A01ALA
Gate  G&F-%F-165-A01ALA      *           %F-165-A01ALA           S%165-A01ALA
Desc  G&F-%F-165-A01ALA Gate event for %F-165-A01ALA
Value %F-165-A01ALA  3.19E-06
Desc  %F-165-A01ALA fire initiating event of 165-A01ALA
Value S%165-A01ALA   0.2
Desc  S%165-A01ALA fire severity event of 165-A01ALA
```

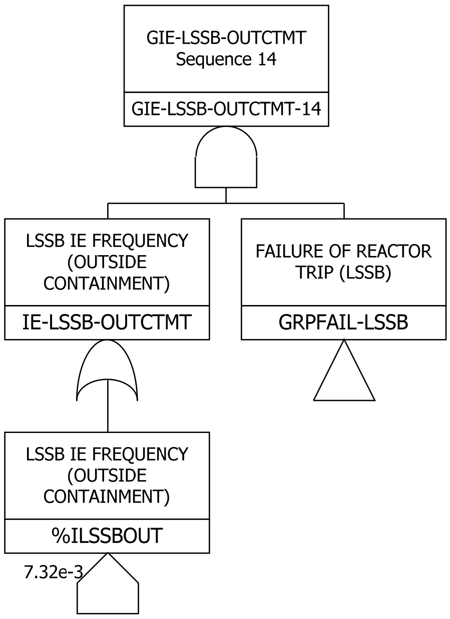


Figure 2. **%ILSSBOUT modeling in Internal PRA for UCN 3**

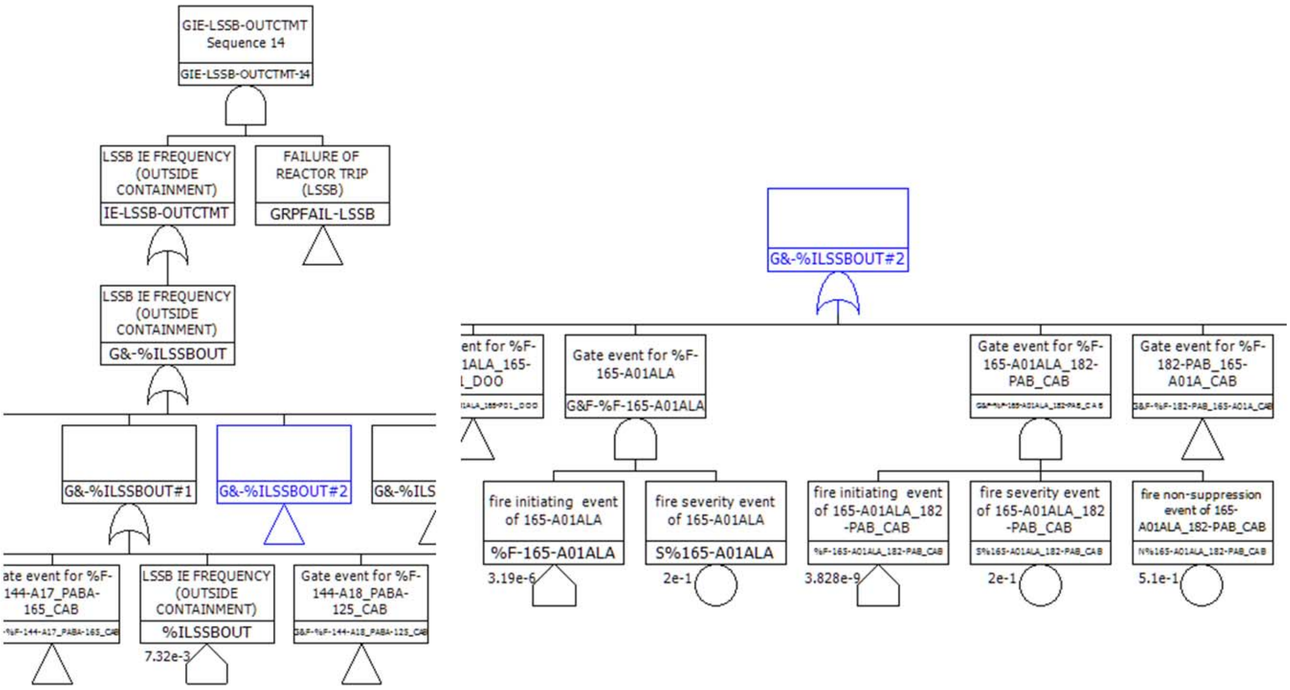


Figure 3. **Automatic modeling of %ILSSBOUT induced by room 165-A01A fire by IPRO-Zone**

Table 6. **Initiating Events for MSO Compartments**

Room	IE	Equip. ID	Raceway type	Required Parts Failure			Conservative Success criteria	SO Prob.	3 min. duration	combined
165-A01A	LSSB-AB	9521V0171	MCT	AC pump circuit	Solenoid VV 1	Solenoid VV 2	AC*SV	1.57E-01 (=0.28*0.56)	2.22E-02 (=0.149*0.149)	3.48E-03 (=1.57E-01 *2.22E-02)
		9521V0172	MCT	AC pump circuit	Solenoid VV 1	Solenoid VV 2	AC*SV	1.57E-01 (=0.28*0.56)	2.22E-02 (=0.149*0.149)	3.48E-03 (=1.57E-01 *2.22E-02)
										6.96E-03

Spurious Operation Scenarios
Knowledge Sharing from the MSO Mapping of Ringhals PWRs

Anna Bjereld
Ringhals AB, Vattenfall, Sweden

Abstract

Fire-induced spurious operation is a probable event in case of fire damage on cables. Therefore, spurious failure modes will soon be considered in fire PRA and safe shutdown analysis for Ringhals PWRs. This paper will summarize an analysis of spurious operations that has been performed in order to identify relevant spurious scenarios to be included in the fault trees, with the following main conclusions: 1) 25 multiple spurious operation scenarios (MSOs) have been identified as applicable and relevant to Ringhals 3 and Ringhals 4. 2) Fire scenarios have been identified that lead to loss of coolant through PORVs and RHR valves, i.e. emergency core cooling might be needed in case of fire. 3) Most spurious scenarios arise in case of fire in the relay rooms, cable rooms and switchgears.

The method used is based on NEI 00-01 Rev. 3, with adjustments to comply with Swedish regulations. This paper describes how by using the method, a fault tree manipulation came up with twelve spurious events to consider (most of them also included in the generic MSO list in NEI 00-01), and how six-factor frequencies of core damage were calculated as a probabilistic input for an expert panel. Though NEI 00-01 provides a risk-informed method, the experience from Ringhals is that conclusions often focus on deterministic aspects.

1. Background

The need to map and analyze spurious scenarios during fire is based on experience from the Browns Ferry fire and subsequent experiments and research about fire phenomena. Former fire protection and early fire PRA considered fire-induced spurious operation to be a low probability event with simultaneous multiple spurious operations (MSOs) considered extremely unlikely. EPRI Cable Testing in 1990, later confirmed by NRC testing, showed that fire-induced spurious operation and MSOs are not unlikely, and should be considered.

This paper intends to share knowledge gained when fire-induced spurious operations were being analysed for the three-loop Westinghouse PWRs Ringhals 3 and Ringhals 4 in 2013.

The Ringhals Nuclear Power Plant is obligated by the Swedish Radiation Safety Authority (SSM) to verify that its four reactors comply with new, modern safety regulations concerning design and construction. The licensee must show that the safety systems are well-separated and that the reactors can be brought to a safe state at any fire, even if all equipment in a fire cell is assumed to fail. In addition, R3 and R4 have undertaken to consider spurious scenarios in their fire analyses.

In the present PRA and safe shutdown[1] (SSD) analysis for the R3 and R4, fire is assumed to cut off any control signals and power supply in the damaged cables, whereby any affected component goes to its de-energized failure mode. A few spurious events, mainly pump trips and breaker actuations, are modelled in the PRA, but none of these events are modelled as fire-induced. The purpose of analysing fire-induced spurious scenarios is to find scenarios that could cause a transient or affect the plant's ability to reach a safe state[2] and use this as input for the fault tree modelling.

2. Method

The analysis follows the methodology described in the Nuclear Energy Institute guidance, NEI 00-01 [3], where fire-induced spurious operation scenarios are treated. Adjustments have been made to comply with Swedish regulations such as the single failure criterion. The methodology in NEI 00-01 comprises multiple spurious operations (MSOs), i.e. fire-induced component failures caused by fire-induced circuit failures including hot shorts. The aim of the methodology is to identify and focus on potential high-risk scenarios.

The major part of the analysis is made up of the MSO mapping, defined by an identification phase, a screening phase and a verification phase.

The purpose of the identification phase is to find all possible spurious scenarios and prepare a comprehensive preliminary MSO list. Scenarios are retrieved from the generic MSO list in NEI 00-01 Appendix G, already identified MSOs from former plant specific Westinghouse analyses, identification of MSOs from flow diagrams, and MSOs from fault tree manipulation (described below under section 5.1 Fault Tree Manipulation).

The purpose of the screening phase is to produce a plant specific draft MSO list by eliminating all scenarios from the preliminary list that are not considered as possible:

- MSOs that are not applicable to R3 and R4
- MSOs that are within the acceptance criteria
- MSOs that are not possible considering system design, circuit properties, cable routing or fire assumptions

Deterministic screening criteria are applied, e.g. *The scenario is not causing a transient or degradation of a safety function*; *The scenario claims more than 4 hot shorts, or hot shorts in a certain order to occur*; *The cables of components are fully separated on different trains and in different fire cells*. Probabilistic screening criteria are considered but not applied (see below under section 5.2 Six-Factor Frequency of Core Damage).

The purpose of the verification phase is to review the plant specific draft MSO list and the screening, and prepare a final plant specific MSO list. The verification is performed by an expert panel with a combined competence that includes fire protection, SSD and PRA, deterministic analysis, operation, systems, cables and circuit design.

---

[1] *To address the requirements on separation, a safe shutdown (SSD) analysis based on the US regulation 10CFR50 Appendix R [1] and the corresponding NUREG Guideline 1778 [2] has been performed for R3 and R4. The method's framework is adjusted for SSM requirements to include additional criteria on single failure, common cause failure, and shutdown operation.*

[2] *Safe state corresponds to assured sub-criticality and a temperature below 100 degrees Celsius in the reactor pressure vessel.*

The outcome of the MSO mapping is a list of scenarios that are considered particularly serious, likely, or in other way important to R3 and R4. The final plant specific MSO list is the basis for conclusions and input for further fault tree analysis.

3. Acceptance Criteria

Considering the total damage of one initiating fire in one firecell, causing all non-fire resistant equipment in the firecell to fail, the following must be demonstrated:

- The fire must not result in a transient that might threaten the fuel integrity or the integrity of the reactor coolant pressure boundary.
- One train of systems necessary to achieve and maintain *hot shutdown* must remain intact (also when any single failure is applied). One train of systems necessary to achieve and maintain *cold shutdown* must remain intact (any single failure is assumed to be repaired within a reasonable time and is thus not applied).

The total damage of the fire includes the MSO mapping and the corresponding safe shutdown analysis results for the firecell.

4. Main Results

25 scenarios have been identified as applicable and relevant to R3 and R4. Most of them are within the acceptance criteria if they occur alone. The combination of these scenarios with other conceivable fire damage in the fire cell, such as inadvertent emergency core cooling or a failed train to a safety function, can nevertheless result in exceeding the acceptance criteria, see Table 1.

A few scenarios have consequences that are currently not modelled in the safe shutdown analysis and where the acceptance criteria are exceeded. This set includes charging pump failure due to failure of volume control tank isolation, refuelling water storage tank drainage via containment spray or containment sump valves, spurious opening of pressurizer relief valves (PORVs), and auxiliary feedwater isolation.

The availability of emergency core cooling has previously not been considered at fire events, since no fire has been identified that could result in loss of coolant or other transient causing low pressurizer level. Fire scenarios have now been identified that lead to loss of coolant through pressurizer PORVs and residual heat removal valves, i.e. emergency core cooling might be needed in case of fire.

Most of the identified scenarios are actuated by fire-induced hot shorts to manoeuvre cables in firecells in the electrical building (relay rooms, cable rooms and switchgears). Fires in certain firecells in the containment, turbine building and auxiliary building can also initiate spurious scenarios, mainly by damage to transmitter cables.

Table 1. **Results per firecell where fire-induced MSOs require further safety functions than what is currently assumed in the SSD analysis for R3 and R4.**

Firecell	**Description of possible MSO transients and total damage at fire**
Relay room A, Cable room A, Switchgear A	A fire in one of these firecells might lead to spurious scenarios that give rise to transients where emergency core cooling or depressurization through PORVs is required. SSD analysis cannot verify availability of emergency core cooling or PORV if a single failure is applied. Acceptance criteria are not shown to be met.

Relay room B, Cable room B, Switchgear B, Electric building, other areas	A fire in one of these firecells might lead to spurious scenarios that give rise to transients where emergency core cooling or depressurization through PORVs is required. Availability of PORV is verified since a fire is only assumed to fail one of three PORVs. SSD analysis cannot verify availability of emergency core cooling if a single failure is applied. Acceptance criteria are not shown to be met.
Containment	A fire in this firecell might lead to spurious scenarios that give rise to transients where emergency core cooling or depressurization through PORVs is required. Availability of emergency core cooling is verified since a fire is not assumed to fail any of the ECCS trains. SSD analysis cannot verify availability of a PORV regardless if a single failure is applied. Acceptance criteria are not shown to be met.
Turbine building, different firecells	A fire in one of these firecells might lead to spurious scenarios that give rise to transients where emergency core cooling or depressurization through PORVs is required. Availability of emergency core cooling and PORV is verified since a fire is not assumed to fail any of the ECCS trains or PORVs. Acceptance criteria are met.
Auxiliary building, different firecells	A fire in one of these firecells might lead to spurious scenarios that give rise to transients where depressurization through PORVs is required. SSD analysis cannot verify availability of a PORV regardless if a single failure is applied. Acceptance criteria are not shown to be met.

5. Discussion on PRA Aspects

Though NEI 00-01 provides a risk-informed method, the Ringhals experience is that it is easier to motivate screening of scenarios or conclusions with deterministic arguments than with probabilistic data. Here follows a presentation and discussion on two subareas where probabilistic methods have been used or considered during the analysis.

5.1 Fault tree manipulation

The purpose of fault tree manipulation is to identify additional spurious scenarios which can lead to core damage in the PRA model during the MSO identification phase, by using methods described in NEI 00-01 Appendix F. The approach is to manipulate the probability of spurious events to be disproportionally high[3]. In this way, combinations including spurious events will top the cutset list of hundreds of thousands of sequences leading to core damage when running consequence analysis cases. Possible scenarios are found among the sequences of high probability and few terms. Such single or multiple events are added to the MSO list.

Spurious operations due to fire are not modelled in the present PRA model for R3 and R4. The models only include occasional spurious signals, i.e. spurious pump stop. Therefore the model was

[3] *The probability for spurious operation due to fire was set to P=0.1. A higher value tends to make the computation time unreasonably long without affecting the result.*

supplemented with spurious errors in order to include fire-induced spurious scenarios[4]. Consequence analysis cases were then run for the initiating events such as loss of feedwater, loss of offsite power and other general transients. Scenarios were added to the MSO list, if 1) The cutset fraction of core damage frequency is high, 2) The number of terms including the initiating event is ≤10, and 3) The number of spurious events is ≤4.

Note that this approach cannot catch all spurious actuations, but only those where the lost function of the component is already modelled in the fault trees. Actuations that might give rise to additional transients are not identified[5].

In conclusion, twelve groups of scenarios were identified in the fault tree manipulation:

- Spurious opening of PORV
- Spurious opening of PORV block valve
- Spurious closure of PORVs
- Spurious closure of pressurizer spray valve
- Injection failure (spurious stop of SI pump, pump room cooling or faulty signal from SI pump oil level transmitter)
- Spurious operation of isolation valves to refuelling water storage tank and/or volume control tank
- Residual heat removal failure (spurious stop of RHR pump, pump room cooling or spurious minflow valve closure)
- Spurious stop of component cooling pump
- Auxiliary feedwater failure (spurious closure of steam valve to turbine-driven pump, spurious stop of electrical pump or pump room cooling)
- Recirculation failure (not considered since not included in the SSD model)
- Spurious stop of hydrotest pump (not considered since not included in the SSD model)
- Spurious valve operation at switch-over to freshwater tank (not considered since not included in the SSD model)

The method did not help to recognize additional scenarios to the R3 and R4 MSO list, since all scenarios identified in the fault tree manipulation are already on the generic MSO list from NEI 00-01.

5.2 Six-factor frequency of core damage

NEI 00-01 Rev. 3 provides a risk-informed method where both deterministic and probabilistic data are presented to an expert panel. The probabilistic data includes the "six-factor frequency of core damage", $F{\cdot}P{\cdot}G{\cdot}S{\cdot}C{\cdot}Z$.

---

[4] *Missing functions and spurious functions are identically modelled after manipulation, i.e. spurious valve closure is modelled equal to a valve that fails to open. This is not believed to impact the result of the fault tree manipulation.*

[5] *As an example, spurious start of a pump causing increase in reactor coolant inventory, or spurious valve opening causing a small loss of coolant accident, will not be identified by using this method.*

The fundament is the product $F \cdot P$ which estimates how probable the MSO is to occur.

F, fire frequency per reactor year for relevant fire cells (values are taken from the PRA model, based on Berry's Method)

P, probability for the MSO to occur in case of fire (typically, 0.3-0.6)

In addition, the following factors might reduce the six-factor frequency further.

G, challenging fire (large, medium or small fire – an expected small fire will mitigate the consequences)

S, fire suppression (automatic, manual or none – reliable fire suppression will mitigate the consequences)

C, conditional core damage probability (redundant shutdown paths – undamaged paths and reliable procedures to reach safe shutdown will mitigate the consequences)

Z, number of vulnerable zones (routing aspects – limited cable routing will reduce the risk of being exposed to a fire)

According to NEI 00-01, MSOs can be screened if the six-factor frequencies fulfil certain criteria, e.g. if the change in core damage frequency (delta-CDF) for each component combination for any fire zone is less than 10^{-7} per reactor year.

The expert panel did not agree that that the six-factor is a consistent estimation of the core damage frequency. Thus, the screening criteria were not applied to the R3 and R4 MSO screening as certain aspects of the approach were questioned. The $F \cdot P \cdot G \cdot S \cdot C \cdot Z$ factor does only reflect probability and not consequence (which means that no difference is made between events of such different dignity as inadvertent pressurizer heater operation and inadvertent PORV opening), thereby constituting quite a blunt instrument to the expert panel.

In addition, the fire frequency, *F*, the number with greatest impact on the six-factor core damage frequency and thus the most important factor in the calculation, can vary in a span between 10^{-2} and 10^{-7}. There is a risk that this number might be underestimated because of a weakness of the method that has been used (Berry's Method). Despite the low probability for spurious scenarios caused by fire in the relay rooms and the cable rooms, MSOs in those firecells should not be screened because of the uncertainty of the factor *F*.

The six-factor frequency has instead been used only as an indicative tool, see Table 2.

Table 2. **Presentation of fire analysis results. The table shows the relation between firecells and the corresponding spurious scenarios that can be induced by a fire in that firecell. The colours represent the probability for each spurious scenario in each firecell by colour-coding the calculated six-factor frequency of core damage. Green boxes indicate an insignificant frequency while red boxes indicate a higher contribution. Empty (white) boxes indicate that the objects included in the scenario have no identified routing dependencies and thus cannot be induced by fire in the related firecell**

	#10 (Spurious isolation of VCT and RWST)	#12 (Spurious opening of VCT outlet valves)	#60 (Spurious closure of charging pump suction)	#53.1 (Spurious opening of RHR booster valves)	#15 (RWST drain down via containment sump)	#16 (RWST drain down via containment spray)	#56d (RWST drain down via containment spray actuation)	#80 (Spurious isolation of PORV block valves)	#56 (Spurious ECCS actuation)	#21 (Spurious start of excess RCS makeup)	#18 (Spurious opening of multiple pressurizer PORVs)	#19 (Spurious opening of PORV and PORV block valve)	#17 (Spurious opening of RHR suction valves)	#88 (Spurious closure of RHR recirculation valves)	#23 (Spurious opening of atmospheric steam dump valves)	#24 (Spurious opening of turbine steam dump valves)	#28 (Spurious closure of AFW pump discharge valves)	#29 (Spurious closure of steam supply valve to AFW pump)	#33a (Spurious start of excess steam flow to steam generator)	#36 (Spurious opening of pressurizer spray valves)	#37 (Spurious operation of pressurizer heaters)	#77 (Spurious closure of auxiliary pressurizer spray valves)	#42 (Spurious CCW valve operation causing flow diversion)	#43 (Spurious closure of salt water system valves)	#54.1 (Spurious stop of HVAC)
Relay room A																									
Cable room A																									
Switchgear A																									
Relay room B																									
Cable room B, firecell 1																									
Cable room B, firecell 2																									
Switchgear B																									
Electric building, other area 1																									
Electric building, other area 2																									
Containment																									
Turbine building, firecell 1																									
Turbine building, firecell 2																									
Turbine building, firecell 3																									
Turbine building, firecell 4																									
Auxiliary building, firecell 1																									
Auxiliary building, firecell 2																									
Auxiliary building, firecell 3																									
Auxiliary building, firecell 4																									

6. Conclusions

The analysis of fire-induced spurious operations for Ringhals 3 and 4 resulted in a list of 25 spurious scenarios where fulfilment of both acceptance criteria cannot be shown.

The first acceptance criterion was that a fire must not result in a transient that can threaten the fuel integrity or the integrity of the reactor coolant pressure boundary. Except for scenarios resulting in LOCA such as inadvertent PORV opening, this criterion is fulfilled for most scenarios if they would appear separately. However, according to the method, the total possible damage in one firecell shall be considered (total damage can include loss of safety functions such as emergency core cooling if safe shutdown analysis cannot verify that it remains intact). With respect to the total fire damage, the major part of the 25 identified scenarios might result in transients that have not been shown to fulfil this acceptance criterion.

The second acceptance criterion was that one train of systems necessary to achieve and maintain *hot shutdown* and *cold shutdown* must remain intact. By adding the spurious scenarios to the safe shutdown model and retrieving results for fault tree analysis of each firecell, it is clear that more than half of the 25 identified scenarios do not meet this criterion and must be addressed. Many separation problems have already been identified in the present safe shutdown analysis and have been addressed, but those conclusions were based on the belief that no fire could ever result in a loss of coolant accident, and that emergency core cooling would thus not be necessary in case of fire. Now it turns out that a train for safety injection is required in case of fire in the electric building. Analyses to show that one train is available have not yet been performed.

The identified areas where spurious scenarios can occur in case of fire include firecells in the electric building, but also containment, firecells in the turbine building and auxiliary building.

Besides the requested conclusions on identified spurious scenarios, there is a finding concerning risk-informed methods. Despite the expressed intention of NEI 00-01 to balance deterministic and probabilistic aspects, the deterministic methods have been assigned a greater importance than the probabilistic methods through this analysis. Generally, it can seem to be more difficult to gain acceptance for probabilistic methods.

References

[1] U.S. Nuclear Regulatory Commission (NRC), Appendix R to 10 CFR Part 50, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

[2] U.S. Nuclear Regulatory Commission (NRC), NUREG-1778, Knowledge Base for Post-Fire Safe-Shutdown Analysis

[3] Nuclear Energy Institute (2011), NEI 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis Revision 3

The Fire on Turbine Generator as Dominant Fire Risk at NPP Dukovany (VVER-440)

Authors L.Kolář, F. Štván
ÚJV Řež, a.s., Czech Republic

Abstract

The fires of the auxiliary systems of turbine generator belong to the most important risk factors for the operation of nuclear power plants. In case of Dukovany nuclear power plant fires on the oil system and turbine fires on the hydrogen-cooled generators are the dominant fire risk scenario. The paper presents a brief description of the expert engineering analysis estimating the risk of fire in the turbine hall in relation with the operation of turbine generators first and second unit, which are located in the common area.

In the paper, dominant accident scenarios at turbine generator auxiliary systems resulting in fire or explosion considered in Probabilistic Safety Assessment (PSA) are presented and the estimation method of the frequency of accident scenarios occurrence is also briefly mentioned including a description of their consequences.

In conclusions, respective effects of the oil fire and hydrogen explosion on the secondary circuit equipment and the structure of machine hall building are described. Positive impacts of the leaking roof of the machine hall on size of the accident scenarios consequences and related operational risk at Dukovany NPP are included.

1. Introduction

For the VVER-440 nuclear power plant the risks induced by turbine generator systems are often more significant than at western types of PWR NPPs, due to the location of some safety-related systems and the overall layout of the turbine hall and the whole plant. The turbine hall is common for two units in case of VVER-440, and there are no walls or fire barriers between the turbine generators and safety relevant equipment. A turbine generator failure may thus affect the operation of many systems, even the operation of the other unit.

The main reason of the high fire risk of the machine hall is the high local fire load, existence of conditions enabling the ignition of flammable materials and presence of components of systems important for safety. The highest fire risk in the turbine building is related to the possible fire of flammable materials of turbo generator (TG) auxiliary systems.

The analysis of fire risk was divided into two phases. The aim of the first phase (analysis of the fire risk of unit 1) [1], [2] was estimation of fire risk caused by fires of solid or liquid materials (electric insulation of cables, oil in TG and electrical feed water pump). In the second phase (analysis of the hydrogen fire and explosion in the machine hall building) [3] the fire and explosion risk analysis in the machine hall building was realized.

Both of the analyses of fire risk of turbine hall in Dukovany NPP were developed in form of expert risk estimation using the EPRI documents (Methods of Quantitative Fire hazard Analysis) [5] without application of computational simulations of fire event process as FEM (Finite Element method).

The analyses of the fire risk always contained the following steps:

- Step 1: Analysis of flammable materials types and fire load analysis in the turbine hall.
- Step 2: Analysis of possible leakages and the dimensions of leakages of flammable materials in the turbine hall.
- Step 3: Analysis of possibilities of fire scenarios initiation in the turbine hall.
- Step 4: Estimation of fire scenarios frequencies in the turbine hall.
- Step 5: Analysis of fire resistance of constructions and components (consequences of selected scenarios).
- Step 6: Analysis of consequences of selected fire scenarios (origin of initiation events, deterioration of unit response).
- Step 7: Estimation of unit operational risk caused by fire scenarios inside the turbine hall.

According to the analyses of fire risk of the Dukovany NPP Unit 1 realized, we can conclude that the essential influence on operational risk value on the unit caused by fire results from the fact that the machine hall is common for two units (fire event at one of the units affects the other one and vice versa) and also from the high leakage of turbine hall building construction (combustion gases venting and venting of possible accidental releases of hydrogen).

2. Turbine Hall and its Importance for Unit Operation Risk

The machine hall belongs to each main production unit (MPU), i.e. it is common for two reactor units, see Figure 1. Each machine hall of a MPU contains four TGs. The machine hall neighbour both the longitudinal auxiliary floor and the transversal auxiliary floor. At the north side of machine hall there are 400 kV and 110 kV out substations including block, tap-changing and reserve transformers belonging to the MPU.

The building of machine hall is large hall with platforms inserted inside. On the platforms all the technological devices and systems are placed. Supporting structure of the building is made of steel and the cladding of building is cellular concrete with second lining made of bricks. Windows of the machine hall building are glassed with armored glass in steel frames. The entrance into the building is possible through door and gate. The trussed roof of the building is made of metal plates, concrete screed and is protected with waterproof roofing made of sarking felt. Ventilation of the building is provided by windows and smoke can be removed by the fire protective device for smoke and heat removal represented by ventilation shutters produced by COLT.

The devices in the machine hall are as follows:

- TGs
- Power supply systems of steam generator - SG (main feed water - MFW, auxiliary feed water - AFW, heat removal - HR)
- Compensation pumps - CP
- Distribution frames of power supply system non-vital
- Circulation cooling Water – CCW
- Etc.

Dimensions of the turbine building: Length: 135.5 m, Width: 42 m, Height: 32.5 m

Location of the turbine building and TGs in turbine hall can be seen in following pictures.

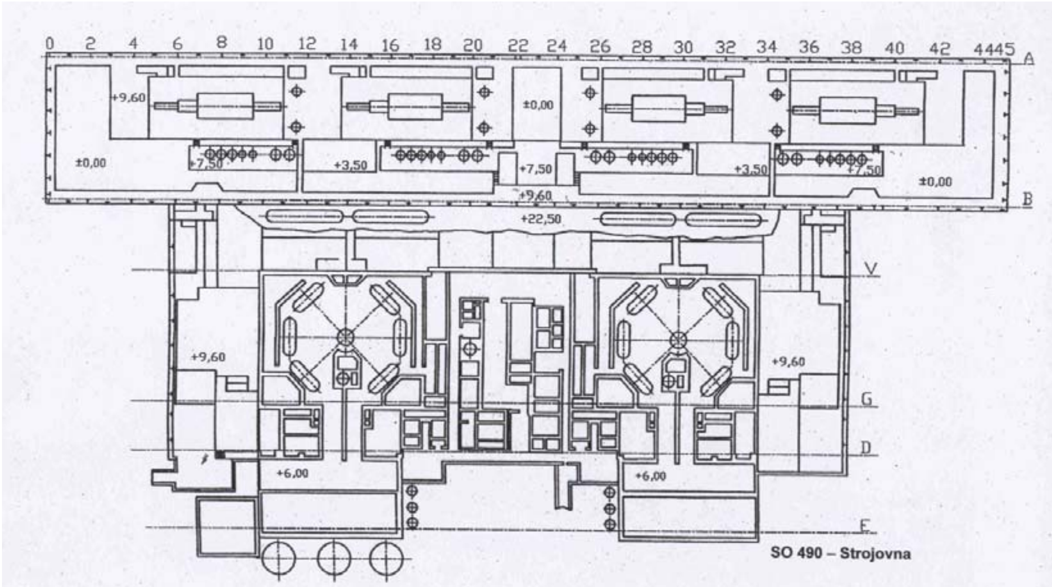


Figure 1. **Four TGs (+9.6m, turbine hall), two reactor units connected with one turbine hall**

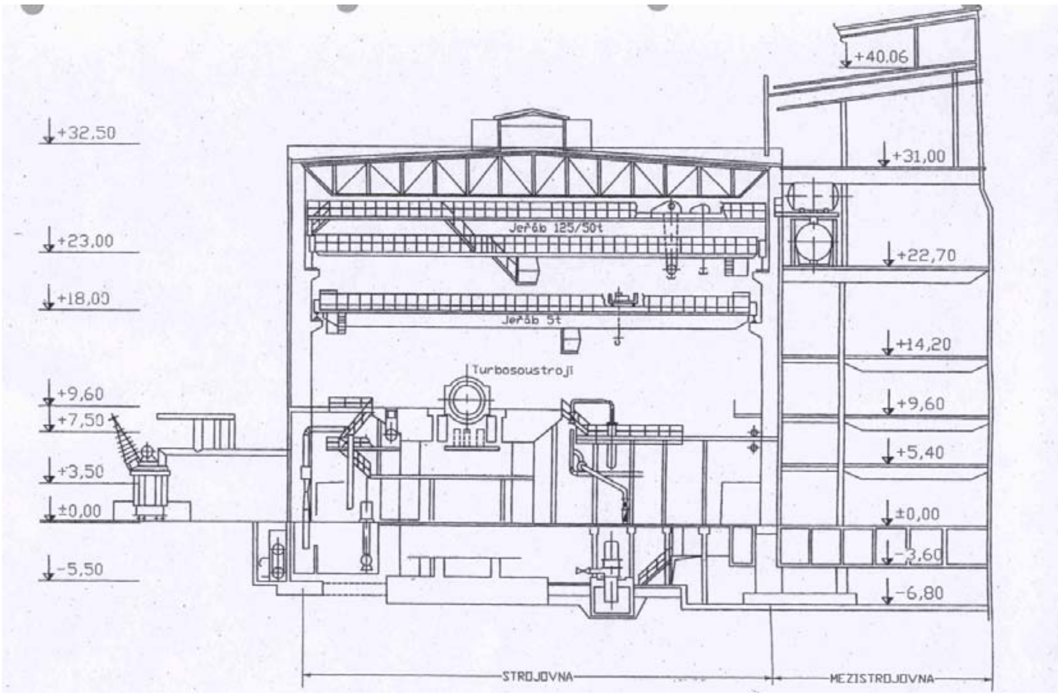


Figure 2. **Location of TG in the turbine hall (+9.6m)**

For the case of VVER-440 reactors the turbine hall and all the systems and devices inside the turbine hall has to be considered as important for risk (design approach). For example from the probabilistic risk assessment we conclude that loss of the turbine hall including all the devices and component inside, the value of CDF risk from internal initiation events (IE) increase to the value about 300 times (risk increase factor RIF=301).

2.1 Main fire safety measures in the turbine hall

Passive fire safety measures:

1. Steel forged frame in the turbine hall are painted with fire-protective painting which prolong the original resistance of 15 minutes to 30 minutes.
2. In the phase of generator design and its hydrogen management standards for device operation in environment with danger of explosions of flammable gases and vapours.
3. Smoking and working with fire is forbidden in the area of generator and its accessories. If welding or other work with fire is needed, control of surrounding air and identification of hydrogen content in the air must be realized before. The works can run only under the control of general foreman and safety engineer. The maximum hydrogen content in surrounding air of turbo alternator or its gas and oil systems can be 1%.
4. No transport by crane above the panels for gas and oil systems control.

Active fire safety measures:

1. Automatic fire detectors EFS (Electrical Fire Signalization) are installed at all potential points of fire event initiation.
2. Analyzers of hydrogen occurrence are installed above the hydrogen system of the generator. Indication is brought out to the control room (CR).
3. In case of long term leak of hydrogen according to the operating procedures and after agreement of the incident commander there is the possibility of opening the ventilation shutters “COLT” placed in the air shaft of the turbine hall. The function of these shutters “COLT” is to ventilate the turbine hall in case of oil fire (outflow of combustion gas) or hydrogen leakages from TG (prevention of explosion).
4. Possible oil leakage at TG will be signalized at CR by pressure change in oil system
5. In cable channels and oil well tubing, there is a shower system controlled by the signal EFS (this system cannot be considered as fire extinguishing system because it doesn’t generate water fog).
6. There is a stable fire extinguishing system installed in the TG oil system area.

3. Main Fire Load (Turbine Hall)

Fire risk of the turbine hall in context of operation of TG is depending on the type and amount of flammable material and also on fire characteristics of the flammable.

Occurrence of flammable materials

Flammable material	Occurrence	Amount / unit (two TGs in one unit)
Oil	Oil system of the turbine.	68 m^3 (2 x 34 m^3)
Hydrogen	Hydrogen system of the turbine.	120 m^3 (2 x 60 m^3)

Considered characteristics of the flammable material

Turbine oil TB32:

Flash point 180°C, Burning point 220°C, Self-ignition point 280°C, According to EPRI: Heat of combustion = 46,4 MJ/kg, Approx. unit heat release rate =1538 KW/m^2, Ideal unit mass loss rate =0,039 kg/s m^2, Density = 760 kg/m^3.

Hydrogen:

Limits of flammability = 4 to 75 % by volume, Combustion heat = 141900 kJ/kg, Combustion heat = 12 760 kJ/m^3 at 20°C, Hydrogen density= 0,00898 kg / m^3, TNT equivalent = 4184 kJ/kg

4. Considered Fire Scenarios for TG (Turbine Hall)

Fire scenarios for TGs can be divided in turbine oil fires and fires or explosions of hydrogen.

4.1 Leakages leading to TG oil fires

The following scenarios leading to significant oil leakages from the oil system in the area of TG were analyzed in terms of TG oil fires:

Considered scenario	Estimation of event frequency [event/year x turbine]	Note
Catastrophic rupture of TG bearing with oil leakage.	about 5 x 10^{-6} [4]	Estimation based on statistics of turbines produced by company Škoda.
Catastrophic rupture of TG bearing with turbine shaft trip (TG like a missile).	on order of 10^{-7} to 10^{-8} [4]	Expert estimation for large turbines.
Significant oil leakage from the oil system (rupture of the piping, leakage).	1.51 x 10^{-3} (modelled in PSA)	Using the ČSN standard for estimation of fire frequency for 1m^2 of turbine hall following with calculation for fire at TG according to EPRI (fire ration in different locations of the turbine hall).
Affection of TG oil system by the broken blade of the turbine Low Pressure (LP) stage (defect of blade material).	3.30 x 10^{-4} [4] (modelled in PSA)	Missile origin: F=5.5 x 10^{-3} Impact of target: P= 6 x 10^{-2} Target damage: P =1
Affection of TG oil system by the broken blade of the turbine Low Pressure (LP) stage (high	On order of 10^{-8}	Analysis of frequency of occurrence of events leading to unloading of TG and

Considered scenario	Estimation of event frequency [event/year x turbine]	Note
revs of turbine)		consequent control failure.

During normal operation hot surface of the piping with hot steam (+256° C) is considered as a constant source of ignition of leaking oil. The transient source of ignition can be represented by electric short-circuit, some maintenance jobs etc. We consider conservatively the probability of ignition P=1. The probability of expanded fire is supposed to be P=0.3 (according to ČSN standards it is taking into account presence of the fire extinguishing system).

For possible fire we presume (considered in PSA):

1. In case of leakage from the oil system we presume the leakage of ½ of overall quantity of oil (17 m3). Discharge of oil through the oil cooler is 2700 l / minute.
2. Leaking oil will drain down the building-in of the turbine hall and at the level of 0.0 m in the turbine hall will create a basin with diameter of 10 m (314 m2).
3. In case of ignition the amount of oil discharged (12920 kg) will according to EPRI documents [4] be on fire with combustion velocity 0.039 kg/s m2. Time to oil burn-up is approximately 18 minutes.

4.2 Leakages leading to fire or hydrogen explosion

For analyses purpose, accidental leakages of hydrogen are represented by all the leakages caused by following causes:

Considered scenario	Estimation of event frequency [event/year x turbine]	Note
Heavy load drops on the piping of hydrogen distribution system in the turbine hall (forbidden transport).	1.8×10^{-7} [3]	Small to medium leakage with small contribution to CDF (small fire radius and small radius of heat radiation).
Heavy load drops on technological devices of the hydrogen system in the turbine hall (forbidden transport).	9.5×10^{-8} [3]	Small to medium leakage with small contribution to CDF (small fire radius and small radius of heat radiation).
Failure of operating crew during the maintenance, leading to ignition of operational leakages of hydrogen at hydrogen system of TG.	5.0×10^{-4} [3]	Formation of small close-oriented fire with small contribution to CDF (small fire radius and small radius of heat radiation).
Trip of TG shaft from the bearing bed with damage of TG hydrogen seals.	4.17×10^{-8} [4]	Output of the analysis of missiles. Small contribution to CDF.
Vibrations of TG leading to destruction of TG hydrogen seals.	6.67×10^{-9}	In case of failure hydrogen seals in TG, hydrogen releases immediately from TG into the turbine hall building.

Note to operational leakages:

During the operation of the NPP Dukovany, there has never been such leakage of hydrogen which would pass the permitted limit of 30 m^3 per 24 hours. Considering these numbers we can calculate that the permitted leakage of hydrogen at one unit is 2.5 m^3 / hour. The real leakage measured is about 1.7 m^3 / hour. Most of the service leakages of hydrogen is led out above the roof of machine hall building, e.g. outside the building.

Previous scenarios are characterized by enough low frequency of occurrence or have small local consequences (small fire radius and small radius of heat radiation). Most of scenarios interesting for risk will be connected to failures of component of hydrogen system or loss of function of the hydrogen sealing in the TG. There are many of these scenarios but unfortunately there are no specific data.

For the purpose of estimation of annual frequency of occurrence of hydrogen fire or explosion interesting from the point of view of risk, the following documents were used as generic sources:

1. EPRI Fire Events Database and Generic Ignition Frequency Model for U.S. Nuclear Power Plants, EPRI, Palo Alto, CA: 2001. 1003111.
2. OECD OECD Fire Incident Records Exchange (database actual to June 2011)
3. IRS Incident Reporting System, IAEA (database actual to June 2011)

For the analysis, releases of hydrogen were divided to small to medium release and large leakages with characteristics described in the following text.

4.2.1 Small and medium leakages

Small and medium leakages include operational and accidental leakages from the hydrogen system of TG where the size of leakage is limited by the largest hydrogen piping (1 inch) and by the maximum pressure of hydrogen in the piping of 600 kPa. This description characterises the maximum leakage of category of small and medium leakages which means 2.39 x 10^{-3} kg hydrogen / second.

In case of leakage ignition

The fires due to small and medium leakages are local and limited in direction (local fire in the area of leakage occurrence with small fire radius and small radius of heat radiation). The risk of these fires is negligible in comparison with the total value of CDF from internal IE.

In case of unfired leakages:

Scenario A (original analysis from 2011, airtight turbine hall)

In scenario A, hydrogen leakages represent the risk of creation of explosive concentration of hydrogen under the roof of the machine hall. According to preliminary analyses, the time interval to necessary ventilation of hydrogen by opening the ventilation shutters “COLT” is 8 minutes. If it wouldn’t be possible to vent the hydrogen accumulated, the consequent explosion will cause destruction of the roof construction of the machine hall (overloading the permitted value of air blast pressure 10 kPa) and the wreckage will fall down on the equipment of the machine hall. This can be calculated by TNT equivalent (explosion cloud yield 10%).

Scenario B (new analysis from 2012, unairtight turbine hall)

In scenario B, it is supposed that gas leakages are vented by the leaks in the roof of turbine hall. Then the gas leakages do not represent any risk for the operation of the unit.

4.2.2 Large leakages

Large leakages include all accidental leakages when all the hydrogen of TG system is released quickly (accidents leading to damage of hydrogen sealing of TG both from hydrogen and air side).

The maximum amount of hydrogen in one TG is 60 m^3 of hydrogen with pressure 300 kPa. The leakage in the machine hall in the atmospheric pressure means about 200 m^3 of hydrogen. Expressed in units of mass, the maximum amount of hydrogen in one TG is 1.8 kg of hydrogen which corresponds to the force of 6.1 kg of TNT (assuming the maximum explosion yield 10%). The speed of hydrogen leakage is so fast that is not possible to count on the reaction of operating crew, i.e. opening of the ventilation shutters “COLT” of the roof construction with consequent venting of accumulated hydrogen. In the analysis, probability of hydrogen ignition and consecutive explosion is assumed as “1”. Early outage of electric power of the crane rails as a prevention of hydrogen ignition in the space under the roof of machine hall is possible only for small and medium hydrogen leakages where the reaction time for operating crew is longer.

In case of early ignition of large leakages:

Leakages can represent the risk of volume fire event of the cloud with hydrogen concentration below or above the interval of explosibility of hydrogen. In case of fire, equipment and devices in the machine hall would be destroyed by the radiation heat flux.

In case of unfired leakages:

Scenario A (original analysis from 2011, airtight turbine hall)

In the area of catastrophic hydrogen leakage, the concentration would be above the explosion limit. The hydrogen accumulates under the roof construction and there is no time to vent it. The following explosion will cause according to the calculation of TNT equivalent destruction of the roof construction of the machine hall (overloading the permitted value of air blast pressure 10 kPa) and the wreckage will fall down on the equipment of the machine hall.

Scenario B (new analysis from 2012, unairtight machine room)

In the area of catastrophic hydrogen leakage, the concentration would be above the explosion limit. The hydrogen released expands quickly in the machine hall building and leaks through leakages in the roof construction. It is important to note that areas with explosive concentration of hydrogen can occur. On the other hand, in this case expected leakages of hydrogen through the roof would decrease the amount of hydrogen accumulated and so the possible explosion would be smaller than in case of hydrogen accumulation of hydrogen under the airtight roof of turbine hall. An explosion of a smaller cloud of hydrogen with the same explosion yield is characteristic with lower pressure at the shock front than the one described in scenario A. The analyses confirmed that the upper limit of components fragilities at pressure of 10 kPa won’t be passed with high probability.

In case of a real accident of catastrophic (immediate) leakage of all the hydrogen from the TG, the possibility of occurrence of small local explosions cannot be excluded. From the point of view of consequences for the unit, the volume fire of the cloud composed of all the hydrogen in TG when all the energy of hydrogen is conversed into radiation heat energy of fire and the damage of components is caused by thermal load.

Estimation of annual frequency of occurrence of conditions of catastrophic fire in turbine hall:

The annual frequency of occurrence of large consequences like hydrogen fire or explosion as a function of operational state of the reactor, derived in terms of the database EPRI [6] is presented in table below:

State of the reactor unit	Basic frequency of hydrogen fire or explosion occurrence [event/year x turbine]	Coefficient of occurrence of large consequences	Frequency of occurrence of large consequences [event/year x turbine]
Full power state	$7{,}7 \times 10^{-3}$	1/12	$\underline{6{,}42 \times 10^{-4}}$
Low power or no-power state	$2{,}5 \times 10^{-3}$	1/2	$\underline{1{,}25 \times 10^{-3}}$

4.3 Airtight – unairtight machine hall transition

At the end of the year 2011, the results of analysis of hydrogen explosion considering the airtight machine hall (during winter, 3/12 year) were unacceptable and the following facts were learned. Neither the windows of the air shaft “VEMA” nor the ventilation shutters “COLT "are equipped with sealing, the windows don’t fit the casing and many leakages occur. Another fact shows that in the space between the windows and the roof of the air shaft, there is a stable air escape along the entire air shaft circumferential. The height of the escape is 15 cm which means for all the air shaft about 420 m (the air escape has 63 m^2).

In the following figures, significant leakages in the roof construction can be seen which eliminates the possibility of hydrogen accumulation under the roof construction.

Figure 4. **Leakages of the windows “VEKA”, no sealing installed**

Figure 5. **Air escape in the roof construction (marked with the ellipse)**

Based on the facts mentioned above, the consideration of possible accumulation of hydrogen under the roof construction of machine hall was changed in the risk analyses. This change significantly influences possible consequences of explosion of the hydrogen-air mixture.

5. Considered Fire Resistance of Components

Fragility of components in turbine hall due to radiation heat flux was defined by the maximum heat resistance of the electrical isolation of cables.

Criteria of cabling failure according to EPRI [5]

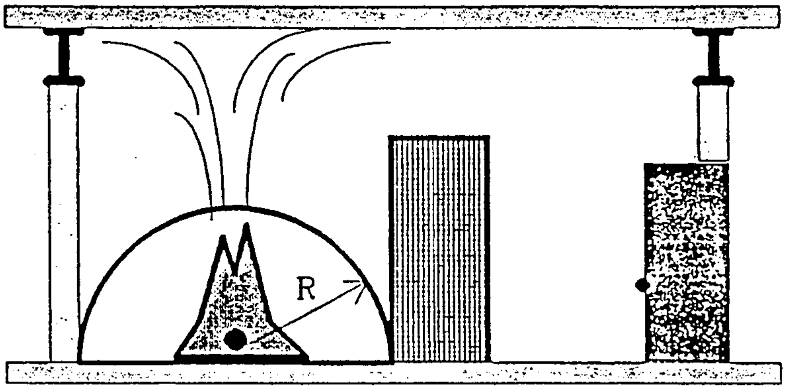
Type of cabling	Temperature of failure	Heat flux leading to failure (q_{CRIT})
PVC cabling qualified according to IEEE-383	370 ^{O}C	10 kW/m^2
PVC cabling not qualified	218 ^{O}C	5 kW/m^2

<u>The resistance of building constructions of the turbine hall against the effects of explosion pressure wave were determined in conservative manner according the pressure design for 10 kPa.</u>

In the following chapters including the method of quantification of radiation heat flux of the flame and of functions of pressure wave generated by hydrogen explosion.

5.1 Radiation heat flux of the flame (combustion)

Calculation of the radiation heat flux in the distance R:



$$q_R = Q_R / \left(4 \times \pi \times R^2\right)$$

q_R	Radiation heat flux as a consequence of fire in distance R from the point of fire [kW/m^2]
Q_R	Exchanged heat energy by heat radiation [kW]
R	Distance R [m]

Calculation of energy exchanged by heat radiation

$$Q_R = Q_{\max} \times X_R$$

Q_R	Total exchanged heat energy [kW]
Q_{max}	Maximum energy released [kW], can be determined from the combustion heat and combustion velocity (reaction rate)
X_R	Radiation coefficient [-], 0.4 used

$$Q_{\max} = q \times A_p \quad \text{[KW]}$$

Input values:

q	Turbine oil	1538 KW / m^2
A_p	Fire surface	314 m^2

Calculation of critical distance from the point of view of failure of cable lines:

$$R_{CRIT} = \sqrt{Q_R / \left(4 \times \pi \times q_{CRIT}\right)}$$

R_{CRIT}	Critical distance [m]
Q_R	Total exchanged heat energy [kW]
q_{CRIT}	Critical heat flux from the point of view of fragility of cable lines [10 kW/m2]

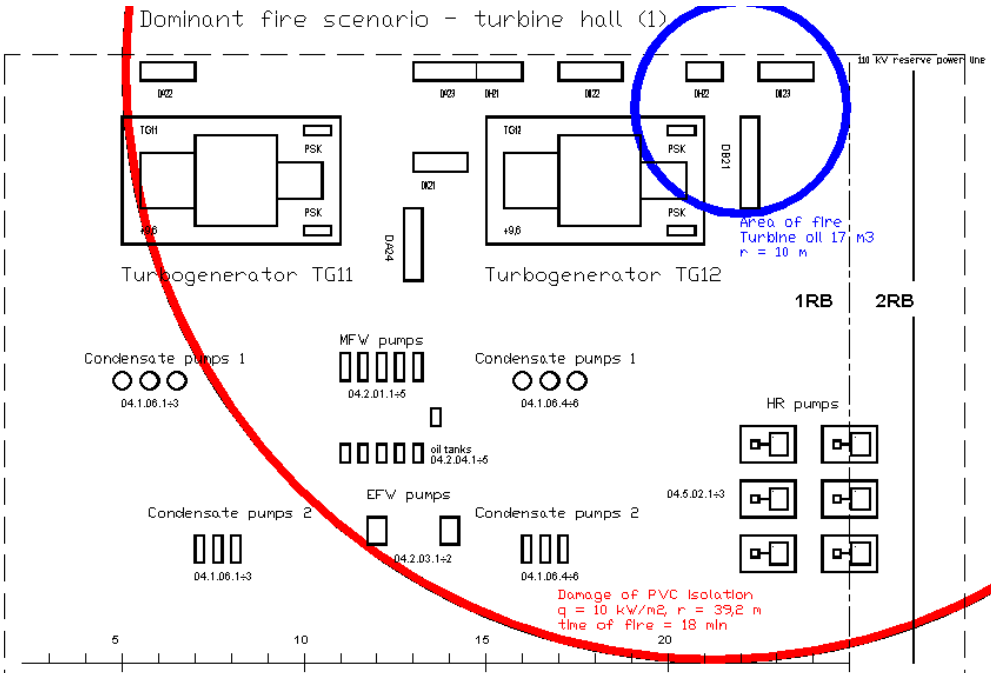


Figure 6. **Range of the destroying effects of radiation heat flux of a large fire of turbine oil at Dukovany NPP, reactor unit 1.**

5.2 Explosion pressure wave

For determination of overpressure at the front of air shock wave that releases to the surroundings we start with the reduced distance. The reduced distance is defined as follows:

$$\bar{R} = \frac{R}{\sqrt[3]{C_W}}$$

$\bar{R}$	Reduced distance from the epicentre of fire [m/kg $^{1/3}$]
R	Distance from the epicentre of explosion [m]
C_W	Equivalent mass of the explosive charge [kg TNT]

where

$$C_W = \alpha_e \times \frac{C_N \times H_N}{H_{TNT}} = \alpha_m \times C_N$$

C_W	Equivalent mass of the explosive charge [kg TNT]
C_N	Mass of the real flammable material [kg H_2]
H_N	Combustion heat of the real flammable material [J/kg]
H_{TNT}	Explosion energy TNT [J/kg], *according to sources 4.18 to 4.65 MJ/kg*
α_e	TNT-equivalent based on energy
α_m	TNT-equivalent based on amount

„TNT- equivalent“ can be considered as a conversion factor which expresses the fraction of heat that is transported to the energy of pressure wave. It means that “TNT-equivalent” represents the efficiency of the conversion process of the chemical energy (combustion heat) to mechanical energy (effect of pressure wave).

For PSA purpose the following algorithm (Makovička D., [8]) was used for calculation of pressure wave parameters.

The maximum overpressure and underpressure at the front of air shock wave and the time of duration can be calculated according to following equations:

$$p_+ = \frac{1{,}07}{\bar{R}^3} - 0{,}1 \quad \text{[MPa]} \quad \text{For} \quad \bar{R} \le 1$$

$$p_+ = \frac{0{,}0932}{\bar{R}} + \frac{0{,}383}{\bar{R}^2} + \frac{1{,}275}{\bar{R}^3} \quad \text{[MPa]} \quad \text{For} \quad 1 < \bar{R} \le 15$$

$$p_- = \frac{0{,}035}{\bar{R}} \quad \text{[MPa]}$$

$$\tau_+ = 1{,}6 \times 10^{-3} \times \sqrt[6]{C_W} \times \sqrt{R} \quad \text{[s]}$$

$$\tau_- = 1{,}6 \times 10^{-2} \times \sqrt[3]{C_W} \quad \text{[s]}$$

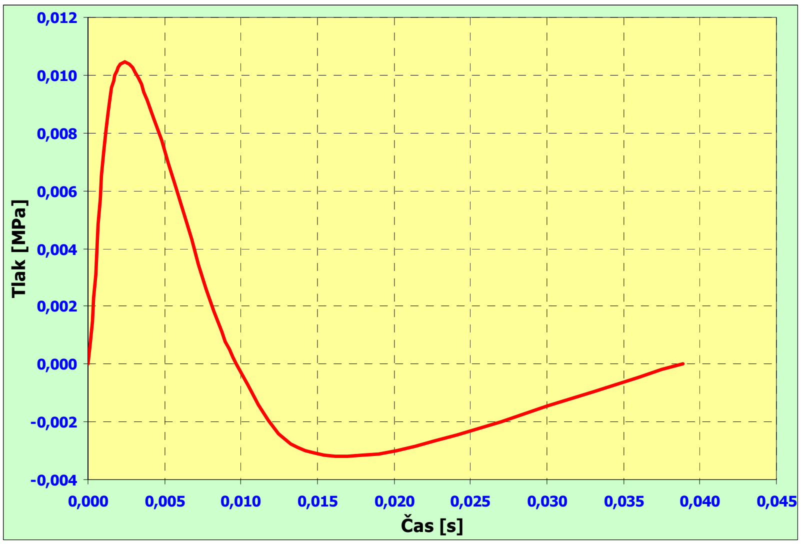


Figure 6. **Time evolution of pressure wave in the distance of 20 m from the epicentre of explosion of 6,1kg TNT (30 m above terrain, ballistic ratio of explosive charge $k_{B=}$ 0)**

Evaluation of the fragility of building constructions and components is realised by comparison of resistance of building constructions and components with pressure values calculated (including time of duration) at the front of air shock wave.

For the analysis of risk due to leakages of hydrogen in turbine hall, the starting consideration is that 10% of hydrogen released enters the explosion reaction.

6. Results of the Risk Analysis of Fire or Explosion in Context of TG

In the following text the main results of the analysis of fire and explosion risk are presented for the case of Dukovany NPP Unit 1 with a view on fire or explosion connected with operation of TG.

6.1 Fire risk for operation of unit 1

The operational risk of Unit 1 (January 2014) from all internal IE: CDF=1.04 x 10^{-5}

The total risk from the fire event at Unit 1 is CDF=1.15 x 10^{-6}, which means 11% of the total risk of operation of Unit 1 comprehending all internal IE.

6.2 Fire risk from TG

The risk of operation of the Unit 1 from the fire of turbine oil or hydrogen fire (eventually explosion) at TG of Unit 1 or Unit 2 is CDF=1.08 x 10^{-6}, which means 94% of the total fire risk and about 10% of the total risk of operation of unit comprehending all internal IE.

The contribution of accident scenarios for full power states of the unit participate on the risk with 90.6%.

The contribution of accident scenarios for low power and shutdown of the unit participate on the risk with 9.4%.

The values presented are composed of contributions of Unit 1 and Unit 2:

Unit 1 (fire at TG11, TG12) contributes with the value CDF=9.55 x 10^{-7} which means 88% of the total value.

Unit 2 (fire at TG21) contributes with the value CDF=1.25 x 10^{-7} which means about 12% of the total value.

6.2.1 Fire of TG oil

Oil fire participates on the total fire risk at operation state of TG by 39.4% and reaches the value of CDF= 4.26 x 10^{-7}.

6.2.2 Fire of TG hydrogen

Hydrogen fire participates on the total fire risk at operation state of TG by 60.6% and reaches the value of CDF= 6.54 x 10^{-7}.

6.3 The effect of air-tightness of turbine hall

In case of airtight turbine hall and possible occurrence of catastrophic explosion with consecutive fall of roof construction of the building, the total hydrogen fire and explosion risk of the machine hall at operation state is about CDF=3.13 x 10^{-6}, which means 25% of the total value of risk caused by all internal IE.

In case of unairtight turbine hall with sufficient ventilation of hydrogen released, there is no danger of explosion with sufficient overpressure at the front of pressure wave (effecting fall of roof construction) and so the total operational risk due to hydrogen fire or explosion in the machine hall can be estimated with the value about CDF = 6.54 x 10^{-7}, which represents 6.3% of the total risk caused by all internal IE.

The tightness of machine hall building has an essential influence on reduction of operational risk of the unit due to fire event.

7. Conclusions

In case of reactors VVER-440, the turbine building has to be considered as a building important for risk (deterministic design approach). Probabilistic risk assessment of unit operation shows that in case of lost of machine hall including all the devices and components in machine hall the value of CDF will increase to about 300 x (RIF=3,01 x 10^{+2}).

This is the reason why large oil or hydrogen fire events on TG systems that disable important components and devices in the turbine hall, have to be take into account with regard to inconsiderable contribution to total operational risk of the unit.

The contribution of fire event related to the operation of TG to the total value of risk caused by all internal IE is about 10% (in absolute values CDF=1.08 x 10^{-6}).

The value of risk presented above is generated by 90% of scenarios for full power unit operation and of 10% of scenarios for low power or non-power unit states which corresponds to the possibility of ignition of oil leakages and presence of hydrogen in TG and its supporting systems.

The fire risk connected with operation of TG is composed of 60% (CDF= 6.54 x 10^{-7}) by hydrogen fire and of 40% (CDF= 4.26 x 10^{-7}) by oil fire.

The essential measure that should be ensured in terms of fire risk is the sufficient ventilation of turbine hall to prevent explosions of large cloud of air-hydrogen mixture. In case of good ventilation the contribution of fire risk to the total operational risk of the unit caused by all internal IE can be decreased by 20%.

References

[1] Kolář L., Probabilistic Assessment of Dukovany NPP (Fire Analysis for 1st unit, R1), 2001

[2] Kolář L., Living PSA, Probabilistic Assessment of Dukovany NPP (Fire Analysis for 1st unit, LPS), 2001

[3] Kolář L., Štván F., Výbuchy H_2 na strojovně, ÚJV Řež, říjen 2011, ÚJV Z 3249 T

[4] Kolář L., Pravděpodobnostní hodnocení rizika událostí typu letících předmětů v objektu strojovny, ÚJV Řež, a.s., srpen 2011, ÚJV Z 3187 T

[5] EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE), 1992

[6] Fire Events Database and Generic Ignition Frequency Model for U.S. Nuclear Power Plants, EPRI, Palo Alto, CA: 2001. 1003111.

[7] Makovička, D., Shock Wave Load of Windows Glass Plate Structure and Hypothesis of Its Failure. In: Structures Under Shock and Impact 1998. Computational Mechanics Publications, WIT Press, p. 43-52, Southampton 1998

[8] Makovička, D., Makovička, D., Zjednodušený výpočet tlakové vlny, http://pvoch.cvut.cz/vypocet_vlny/, Praha 2009

Updating of the Fire PRA of the Olkiluoto NPP Units 1 and 2

Lasse Tunturivuori
Teollisuuden Voima Oyj (TVO), Eurajoki, Finland

Abstract

The probabilistic fire risk assessment (fire PRA) of the Olkiluoto 1 and 2 NPP units has been updated in the beginning of the year 2011. In the updated version, the fire frequency of each component group is evaluated using a Bayesian approach with NUREG/CR-6850 data as a prior and using historical fire events at the NPP as evidence.

The resulting total fire frequency estimate in the PRA model of the units is 10 % higher than the historical fire frequency of the NPP units. However, the fire frequency estimate for certain rooms changed by orders of magnitude due to the transition to the usage of component based fire frequencies. Especially, the fire frequency estimate of rooms housing components with exceptionally high fire frequency, like main feed water pumps, large amount of electric equipment, like relay rooms and rooms with hydrogen systems increased in the updated evaluation.

In the analysis, the safety-related components are mapped to their locations. Furthermore, the cable routing database is used in order to locate the power and control cables of the safety-related components. Also the locations of cables transmitting measuring data to the reactor protection system are assessed. The resulting fire scenarios with similar consequences are grouped into tens of initial events using conservative assumptions.

The update of the fire PRA increased the CDF by 16 %. After the update, the CDF originating from fire-related initiating events constitutes of 20 % of the total CDF. The fire PRA demonstrates that in a high-redundant NPP unit, such as the Olkiluoto 1 and 2 NPP units, which has four redundant divisions, each of 50 % capacity, a fire affecting two redundancies increases the conditional core damage probability by a factor of ten compared to a scenario in which only one redundancy is affected by the fire, demonstrating the significance of physical separation between safety-significant equipment and active fire suppression systems in rooms housing two redundancies.

1. Introduction

Olkiluoto NPP units 1 and 2 are boiling water reactors (BWR) built by ASEA-ATOM in the late 70's and early 80's. Both units' safety systems are divided into four redundant subsystems (divisions), each of which has a capacity of 50 %. Thus, two of the divisions must operate in order to successfully perform a given safety function. The components of the divisions are divided by fire compartmentalization. The exception is cables, of which maximum of two divisions may be present in the same fire compartment. In this case, the cables belonging to different divisions are divided by distance. Such fire compartments are always equipped with active fire suppression systems, usually sprinkler systems.

The first probabilistic fire risk assessment (fire PRA) of Olkiluoto 1 and 2 was finished in 1991. It has been updated several times, the latest in the beginning of the year 2011. In the updated version, one major modification is the estimation of room-based fire frequencies using component based fire frequencies. In the implementation of the fire PRA, the NUREG/CR-6850 report [1] has been used as reference.

2. Implementation of the Fire PRA

2.1. Scope of the Fire PRA and partitioning of the plant into physical analysis units

Internal fires, such as fires in the reactor building, turbine building, auxiliary buildings, control building and the waste building, are in the scope of the fire PRA. Further, also the fire water building and the water processing building are in the scope, since they contain equipment which is credited in the PRA model. However, external fires are outside the scope of the fire PRA and they treated in the context of external hazards. The included buildings contain about 1800 rooms. Some rooms are divided into areas having their own room codes, but such rooms are treated as one single room in the analysis. Rooms are part of a fire compartment, which can withhold a fire at least 60 minutes. In one fire compartment there are components and cables belonging to maximum of two divisions. The NPP units are physically divided so that two divisions are located in the southern and western part of the unit and the two other divisions are located in the northern and eastern parts. At the Olkiluoto NPP, there is an on-site fire brigade, which is manned around the clock. All rooms in the plant unit are equipped with fire detectors. In the analysis, the basic assumption is that a fire in a room fails all components and cables in the room, but the fire brigade will be able to withhold the fire inside the room and thus prevent spreading to other rooms.

2.2. Estimation of room-based fire frequencies

A very important part of a fire PRA is the estimation of fire frequencies. In the original implementation of the fire PRA of Olkiluoto 1 and 2, the room-based frequencies were based on a few parameters evaluated with engineering judgement. In the update, the room-based fire frequencies are estimated with a method based on component based fire frequencies.

2.2.1. Old method

A crude but popular method to assess room-based fire frequency estimates is based on NUREG/CR-0654 [2], often referred to as Berry's method. In the implementation used for Olkiluoto 1 and 2, a number of parameters related to human presence, mechanical equipment and electrical equipment was used. Also judgment about the possibility to extinguish an incipient fire (pilot fire) with portable fire extinguishers and the amount of fire load was used. Since the room-based fire frequencies were normalized so that the sum of the room-based fire frequencies equaled to the total fire frequency of the plant unit, the room-based fire frequency for a typical room or compartment would be of the right range. However, the parameter values were based only on engineering judgment without knowledge of fire frequency of specific equipment.

2.2.2. Method based on NUREG/CR-6850

In the updated fire PRA, the room-based fire frequency estimation is based on dividing fire sources into separate groups, or fire bins. Most of these bins consist of countable components, such as pumps, motors, electrical cabinets, etc. Also self-ignition of cables, hydrogen piping, etc., is included. Transient fires, such as trash bin fires, fires induced from hot-work (e.g. welding and grinding) and other human activity are included as well. NUREG/CR-6850 [1] reports fire frequency estimates on a plant unit basis. Inherently to this method it is assumed that all NPP units have the same amount of components, such as valves, pumps, etc.

The component fire source listing, including their locations, is retrieved from the component database and the amount of cable present in the rooms is estimated using the cable pulling database. A walkdown was made during the update of the fire PRA. The walkdown was performed by a fire PRA modeler and a fire systems expert with fire brigade experience. In the walkdown, primarily the locations of the component fire sources were confirmed and influence factors for transient fires, like occupancy of the room, amount of potentially ignition-inducing maintenance activities performed in the room, and presence of flammable gases and liquids, were estimated.

The information about the number of component fire sources, the influence factors of transient fire sources and the total length of cables in the room are used in calculating the room-based fire frequency estimates. The component based fire frequencies are calculated using a Bayesian approach. The fire frequencies presented in [1] are used as a prior distribution and historical fire events of Olkiluoto 1 and 2 are used as evidence. The resulting posterior distribution is the fire frequency used in the analysis. The fire frequency estimates of the fire sources are shown in Table 1. In the choice of component groups and for guidance of component counting, advice from FAQ's presented in Supplement 1 of NUREG/CR-6850 [3] has been used. For estimation of the prior distribution, events listed in [1] are used, when available. For the busbar fire frequencies, fire events from FAQ 07-0035 (cf. [3]) are used, conservatively using the same amount of reactor years, as for other fire sources. However, a gamma distribution is chosen for the prior distribution instead of the proposed log-normal distribution. The calculation of room-specific fire frequency estimates of some rooms is shown in Table 2.

Table 1. **Calculation of component fire frequency estimates (Power operation). The fire frequency estimates are gamma distributions.**

Fire source	Prior distribution			Plant-specific data		Posterior distribution		
	α	β	Mean	N	T	α	β	Mean
Batteries	1.5	2486	6.03E-04	0	54.7	1.5	2540.7	5.90E-04
Main control board	6	2486	2.41E-03	0	54.7	6	2540.7	2.36E-03
Diesel generators	50	2486	2.01E-02	3	54.7	53	2540.7	2.09E-02
Air compressors	5.5	2486	2.21E-03	0	54.7	5.5	2540.7	2.16E-03
Battery chargers	4.5	2486	1.81E-03	0	54.7	4.5	2540.7	1.77E-03
Cable fires caused by cutting and welding	7	1674	4.18E-03	0	54.7	7	1728.7	4.05E-03
Cable run, self-ignited	12	2486	4.83E-03	1	54.7	13	2540.7	5.12E-03
Dryers	6	2486	2.41E-03	0	54.7	6	2540.7	2.36E-03
Electric motors	14.2	2486	5.71E-03	1	54.7	15.2	2540.7	5.98E-03
Electric cabinets - no HEAF*	109.5	2486	4.40E-02	1	54.7	110.5	2540.7	4.35E-02
Electric cabinets - HEAF*	4	2486	1.61E-03	1	54.7	5	2540.7	1.97E-03
Hydrogen fires	10	2486	4.02E-03	1	54.7	11	2540.7	4.33E-03

Junction boxes	3.5	2486	1.41E-03	0	54.7	3.5	2540.7	1.38E-03
Hydrogen recombinators	26	585	4.44E-02	0	54.7	26	639.7	4.06E-02
Pumps	52.5	2486	2.11E-02	0	54.7	52.5	2540.7	2.07E-02
Transformers, dry	23.5	2486	9.45E-03	1	54.7	24.5	2540.7	9.64E-03
Transient fires caused by cutting and welding	33.4	1674	2.00E-02	0	54.7	33.4	1728.7	1.93E-02
Transient fires	29.9	1674	1.79E-02	0	54.7	29.9	1728.7	1.73E-02
Ventilation systems	16.5	2486	6.64E-03	2	54.7	18.5	2540.7	7.28E-03
Yard transformers, catastrophic	10.5	1674	6.27E-03	0	54.7	10.5	1728.7	6.07E-03
Yard transformers, non-catastrophic	22	1674	1.31E-02	0	54.7	22	1728.7	1.27E-02
Transformer yard (others)	3.5	1674	2.09E-03	0	54.7	3.5	1728.7	2.02E-03
Boilers	2.5	2486	1.01E-03	0	54.7	2.5	2540.7	9.84E-04
Main feedwater pumps	16	1674	9.56E-03	0	54.7	16	1728.7	9.26E-03
Turbine generator excitor	7	1674	4.18E-03	0	54.7	7	1728.7	4.05E-03
Turbine generator oil	16	1674	9.56E-03	2	54.7	18	1728.7	1.04E-02
Turbine generator busbar	3.5	1674	2.09E-03	0	54.7	3.5	1728.7	2.02E-03
Diesel generator busbars	7.5	2486	3.02E-03	0	54.7	7.5	2540.7	2.95E-03

*HEAF: High energy arcing fault

2.3. Selection of safety-significant components and cables

The PRA model is used in order to find components that perform mitigating functions after an initiating event and that may be affected by a fire. Initiating event frequencies are evaluated in the PRA model using historical evidence, and thus component failures only leading to an initiating event are not included in the PRA model. Such components, which may be affected by a fire, are identified using the final safety analysis reports (FSAR), piping and instrumentation diagrams and electrical diagrams. Further, measurement devices and cables sending signals to the reactor protection system (RPS) are analyzed as well. The identified safety-significant components are listed together with the cabinets providing power to the component and the cabinets, where their control is performed. This information is used in order to identify the cables, which provide power and control signals to the safety-significant components. The power sources of the busbars to which the power providing cabinets are connected are identified together with the power cables.

The cable pulling database is a very valuable tool in order to identify the locations of the safety-important cables. The database includes the information of the location of the endpoints of segments of cable raceways. With this information and by using cable routing layouts, the locations of the cables can be identified with good accuracy.

Table 2. **Calculation of room-based fire frequency estimates (Power operation): Some examples**

Room	Fire source	Weighting factor		Fire frequency	
		Unit	Room	Unit	Room
Relay room	Junction boxes	2579	49	1.38E-03	2.62E-05
	Electric cabinets	852	144	4.35E-02	7.35E-03
	Transient fires	5126	5	1.73E-02	1.69E-05
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				7.41E-03
Transformer yard	Generator busbar	9	1	2.02E-03	2.25E-04
	Cable run	2632216	260	5.12E-03	5.05E-07
	Junction boxes	2579	2	1.38E-03	1.07E-06
	Transformer - non-catastrophic fires	5	1	1.27E-02	2.55E-03
	Transformer - catastrophic fires	5	1	6.07E-03	1.21E-03
	Transformer yard - others	5	1	2.02E-03	4.05E-04
	Transient fires	5126	3	1.73E-02	1.01E-05
	Cable fires caused by cutting and welding	7146900	260	4.05E-03	1.47E-07
	Transient fires caused by cutting and welding	1571	1	1.93E-02	1.23E-05
	Total				4.41E-03
Pump room	Air compressors	16	1	2.16E-03	1.35E-04
	Cable run	2632216	2157	5.12E-03	4.19E-06
	Junction boxes	2579	9	1.38E-03	4.81E-06
	Pumps	193	10	2.07E-02	1.07E-03
	Electric cabinets	852	1	4.35E-02	5.10E-05
	Transient fires	5126	6	1.73E-02	2.02E-05
	Cable fires caused by cutting and welding	7146900	2186	4.05E-03	1.24E-06
	Transient fires caused by cutting and welding	1571	2	1.93E-02	2.46E-05
	Total				1.31E-03

From the list of safety-important components, power sources and cables, a list of safety-important rooms is compiled. Since initiating event frequency estimates in the PRA model are based on historical events, there is no need to consider fires that only lead to an initiating event. In the fire PRA, fire scenarios that may lead to an initiating event together with the failing of a component performing a mitigating function or fires that may affect a component in such a way that the plant has to be shutdown according to the technical specifications, are analyzed.

2.4. Consequences of fires

Rooms, where the fire probability has historically been judged to be high, or have high fire load, are equipped with active fire suppression systems. The majority of cable rooms are equipped with sprinklers, especially cable rooms, where cables supplying two redundant divisions are present. There exist, however, normal corridors where cables are routed and which are not equipped with active fire suppression systems.

In the fire PRA model, the basic assumption is that all components and power cables in the room involved by the fire fail. It is assumed that the fire spreads to control and instrumentation cables with 20 % probability. However, if a cable room is equipped with an active fire suppression system such as sprinklers, it is assumed that the fire fails components and cables in one subsystem. If the suppression system is deemed satisfactory, it is assumed that it suppresses the fire so that it does not spread to another subsystem with 90 % probability. However, if the separation of the subsystems is deemed unsatisfactory, it is assumed that the fire spreads with 90 % probability. A fire simulation of a cable tunnel in the reactor buildings of Olkiluoto 1 and 2 has been performed [4], which concluded that if the sprinkler system is activated, it prevented the fire from failing cables in the other divisions in all analyzed cases. If the sprinkler system did not activate, the fire failed or spread to the power cables in the other division with 100 % probability and to the control and instrumentation cables in the other division with 60 % probability.

In electrical rooms, it is assumed that a fire fails connections to components in one subdivision and if there is more than one subdivision in the room, which is the case for relay rooms and power systems supplying lower safety class components, the fire spreads with 10 % probability to the other subdivision, cf. Figure 1. Full scale fire experiments on electrical cabinets have been made [5], where it has been concluded, that fires in an electrical cabinet spread very slowly to adjacent cabinets if they are connected. Spreading to a cabinet separated by distance did not occur.

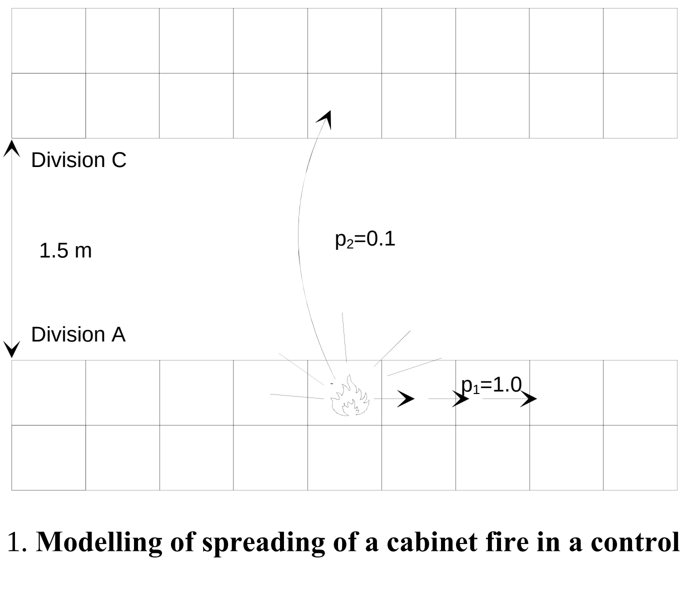


Figure 1. **Modelling of spreading of a cabinet fire in a control room**

The signals of the reactor protection system (RPS) are such that if power is lost, the RPS is launched. Also many solenoid operated valves (SOV) change state if power is lost. In these cases, a hot short in the cable is postulated if this results in a more unsafe state than if the power of the cable is lost.

Fire scenarios are grouped into fire initiating events based on the combination of failures of mitigating functions due to the fire. Thus, instead of modeling hundreds of fire scenarios, just a few tens of fire initiating events are modeled in the PRA model.

Fires affecting the safe shutdown of the reactor, such as failure of the control rod drives, reactor scram system, emergency boron system and shutdown of the main recirculation pumps, are modeled into the fire initiating events. The failure of these functions is modeled in the PRA model with a probability equal to the relative frequency of fire scenarios involving these failures in the initiating event group.

3. Results

3.1. Changes in fire frequencies

In the old method, the fire frequencies of the rooms were normalized so that the estimated total fire frequency equaled the historical fire frequency of Olkiluoto 1 and 2. Contrary to the old method, the new estimates include data from other plants as well. The change of method of estimating fire frequencies increased the estimated total fire frequency of the units Olkiluoto 1 and 2 by 10 % compared to the total fire frequency estimate of the unit using the old method. Furthermore, the room-specific fire frequencies changed for some rooms rather dramatically. This is also the case for some rooms with the highest fire frequency estimates. (cf. Figure 2)

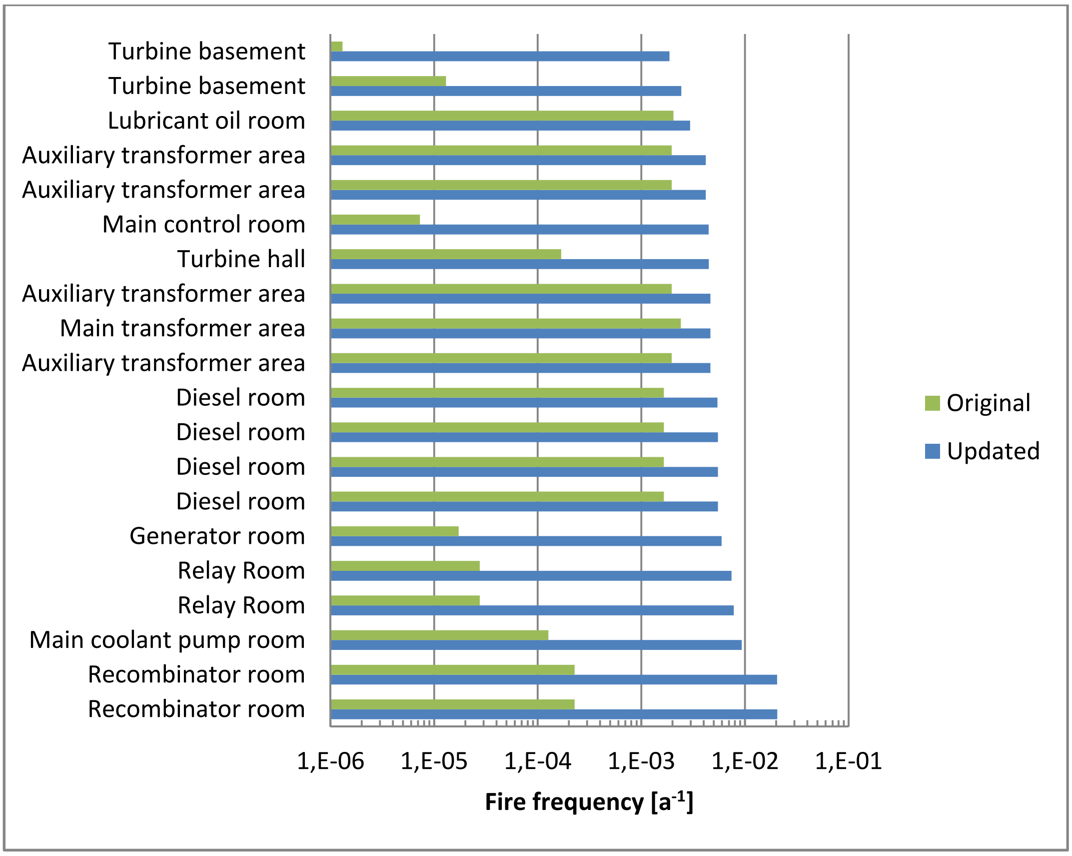


Figure 2. **20 rooms with the highest room-specific fire frequency estimates calculated with the updated method and comparison to the original estimates.**

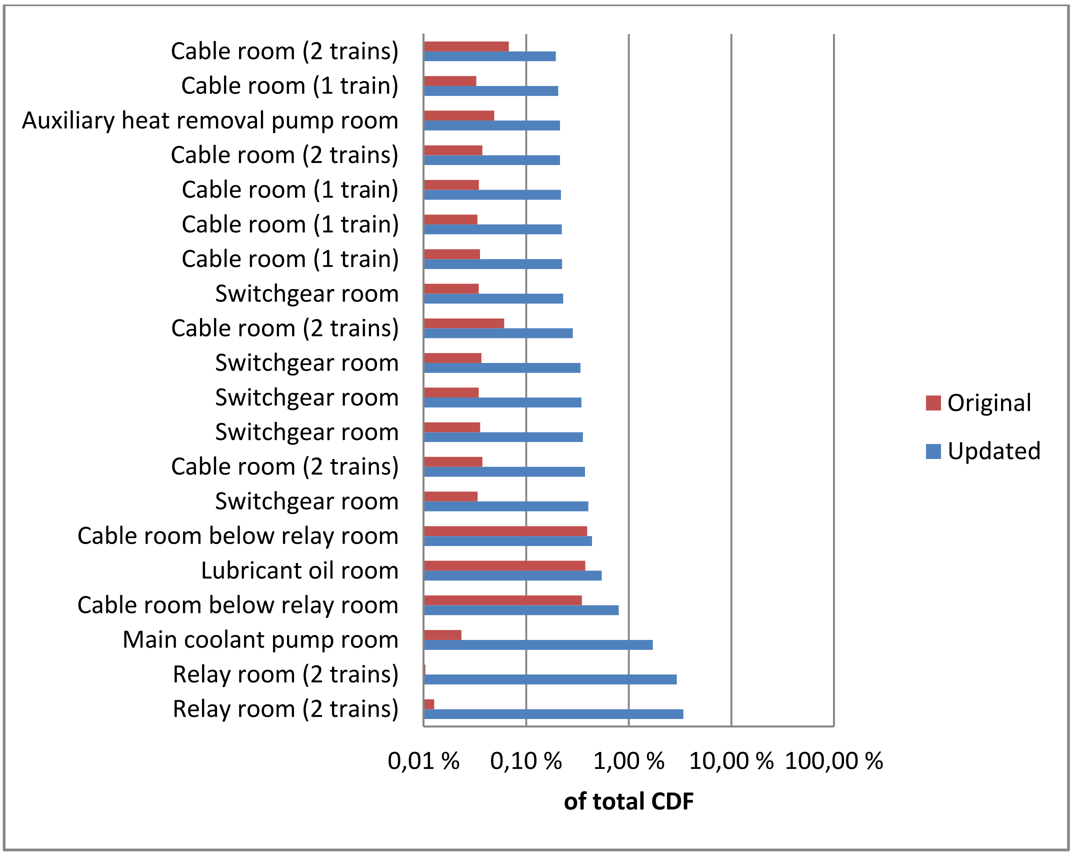


Figure 3. **20 rooms with the highest room-specific core damage frequency estimates resulting from using the updated fire frequency estimates and comparison the core damage frequencies using the original estimates.**

Table 3**. Fire initiating events modelled in the PRA model (Power operation): Some Examples**

Initiating event	Description	Consequence	Birnbaum measure
FIR-01/TP/A	Loss of containment spray, LPSI and HPSI systems, division A	Shutdown of reactor required by TechSpecs	5.70E-07
FIR-02/TP/AC	Loss of containment spray, LPSI and HPSI systems, divisions A and C and RHR system	Shutdown of reactor required by TechSpecs	5.22E-07
FIR-03/TF/BD	Loss of supply to divisions B and D	Loss of feed water transient	2.56E-04
FIR-07/TF	Loss of feed water, unrecoverable	Loss of feed water transient	4.54E-05
FIR-12/TF/B	Loss of supply to division B	Loss of feed water transient	5.72E-05
FIR-12/TF/D	Loss of supply to division D	Loss of feed water transient	5.82E-05
FIR-15/TE	Loss of off-site power, unrecoverable	Loss of off-site power	1.11E-04
FIR-24/TF/BD	Loss of supply to divisions B and D and spurious opening of reactor PRV's	Loss of feed water transient	3.84E-04

The update of the estimated initial event frequencies increased the core damage frequency estimate by 18 %. The increase is, thus, significant but not dramatic. In Figure 3, the rooms with the highest room-specific core damage frequency estimates are shown. The majority of the increase comes from the increased core damage frequency estimate of the relay room fires. Table 3 shows some examples of fire initiating events modeled in the PRA model together with each fire initiating event's Birnbaum measure. The Birnbaum measure expresses the estimated core damage probability with the condition that the fire occurs in the extent that it has been assumed in the modeling of the fire initiating event. It is noticed, that fires just leading to a requirement to shut down the plant have a small impact on the core damage frequency (CDF) estimate. Fires leading to loss of feed water or off-site power, possibly together with failure of equipment belonging to one division, has a higher impact on the CDF estimate, still being reasonably small. If a fire leads to loss of feed water together with loss of two divisions, the CDF estimate increases by an order of magnitude.

4. Discussion

The resulting total fire frequency of the fire PRA model is 10 % higher than the historical fire frequency of the Olkiluoto 1 and 2. This increase adds some conservativeness to the results. The fire frequencies reported in NUREG/CR-6850 [1] are estimated from historical fire events in NPP's in the United States. Due to different design bases between the NPP fleet in the United States and Olkiluoto 1 and 2, the number of components in NPP units in the United States and in Olkiluoto 1 and 2 differ. However, no information on the number of components in the source population is publicly available, which makes it difficult to scale the fire frequencies.

It is clearly noticed that the fire frequency in certain rooms changed by orders of magnitude due to the transition to the usage of component based fire frequencies. Especially, the fire frequency of rooms housing components with exceptionally high fire frequency, like main feed water pumps, large amount of electric equipment, like relay rooms, and rooms with hydrogen systems increased in the updated estimation. This change can at least in part be ascribed to the crude method used in the old implementation in which just a few parameters were used to estimate the final room-based fire frequency. The numbers of parameters were just too few and their bounds limited in order to take exceptionally high (or low) fire frequencies. However, also the choice of component groups will have an effect on the results, especially if components with low fire frequency are joined with the same group as components with a high fire frequency.

The benefit from using component based fire frequencies is not to get plant-wide fire frequencies. That data is readily available for any NPP unit with a long operating history. The importance stems from the need of allocating the total unit-wide fire frequency into room-specific, or depending on the modelling requirements, into component-specific fire frequencies, which can only be estimated to an acceptable degree of accuracy if large statistics are available. The amount of components in different NPP units is of same order of magnitude world-wide, which limits the uncertainty of the fire frequency estimates due to the lack of knowledge of the plant-wide number of components in the statistics in Ref. 1. The component-specific fire frequency estimates are also affected by the choices the analyst has to make when deciding about the criteria used in counting the numbers of fire sources. The criteria given in Ref. 1 together with the FAQ's in Ref. 3 are rather general, giving the analyst a great deal of freedom in setting up analysis-specific criteria.

In the fire PRA of Olkiluoto 1 and 2, the fire frequencies of the main coolant pump room and the relay rooms are very conservative due to very crude fire modelling. It is assumed, that a fire in one main coolant pump automatically spreads to the other three pumps, even though the pumps are protected by sprinkler systems, which effectively inhibit the spreading of the fire to their vicinity. As for the relay rooms, the relay cabinets are grouped into rows consisting of relays belonging to one division. Divisions are separated by distance so that every second row belongs to the same division. Cabinet fire experiments [5] show that the cabinet itself effectively delimit the fire to the cabinet, spreading very slowly to the adjacent cabinets. Therefore, it is expected that detailed modelling of these rooms would decrease the core damage frequency estimate significantly. From Figure 3 it can be deduced that the contribution originating from fires in the relay rooms to the total CDF estimate is 8 % of the total CDF.

Further, it is noticed in Table 3 that fires isolated to one division have only small impact on reactor safety, whereas fires affecting two divisions increase the core damage frequency estimate by an order of magnitude. This can be seen when comparing the fire initiating events FIR-12/TF/B and FIR-12/TF/D with the fire initiating events FIR-03/TF/BD and FIR-24/TF/BD. This exhibits the importance of physical separation of safety-significant equipment. Further, the importance of fire suppression systems in rooms where the fire may fail two divisions is therefore high.

5. Future Work

As discussed above, the results of the main coolant pump room and relay rooms give very conservative CDF estimates. Detailed modelling of fire scenarios in these rooms would decrease the contribution to the CDF estimate originating from fires in these rooms significantly.

6. Conclusions

The probabilistic fire risk assessment (fire PRA) of the Olkiluoto 1 and 2 NPP units has been updated in the beginning of the year 2011. In the updated version, the fire frequency of each component group is evaluated using a Bayesian approach. Data from NUREG/CR-6850 [1] is used as a prior and the historical events of Olkiluoto 1 and 2 are used as evidence.

The resulting total fire frequency in the PRA model of the units is 10 % higher than the historical fire frequency of the NPP unit. However, the fire frequency in certain rooms changed by orders of magnitude due to the transition to the usage of component based fire frequencies. Especially, the fire frequency of rooms housing components with exceptionally high fire frequency, like main feed water pumps, large amount of electric equipment, like relay rooms, and rooms with hydrogen systems increased in the updated evaluation.

The update of the fire PRA increased the CDF by 18 %. After the update, the CDF originating from fire-related initiating events constitutes of 20 % of the total CDF. The fire PRA demonstrates that in a high-redundant NPP unit, such as the Olkiluoto 1 and 2 NPP units, a fire affecting safety systems in only one division has reasonably small consequences to reactor safety. A fire affecting two divisions increases the conditional core damage probability by an order of magnitude. This demonstrates the importance of adequate physical separation between redundant safety-significant components and cables and of effective and reliable active fire suppression systems in rooms, where multiple redundant safety-significant components and cables are present.

References

[1] EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities. EPRI TR-1011989 and NUREG/CR-6850, Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, 2005.

[2] Berry D and Minor E. Nuclear Power Plant Fire Protection - Fire-Hazards Analysis (Subsystems Study Task 4). NUREG/CR-0654, Sandia Laboratories, Albuquerque, NM, 1979.

[3] EPRI/NRC-RES Fire Probabilistic Risk Assessment Methods Enhancements. EPRI 1019259 and NUREG/CR-6850 Supplement 1, Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, 2010.

[4] Matala A and Hostikka S. Probabilistic Simulations of Cable Fires in a Cable Tunnel. VTT Research Report VTT-R-00836-10, VTT Technical Research Centre of Finland, Espoo, Finland, 2010; Matala A and Hostikka S. Probabilistic Simulation of Cable Performance and Water Based Protection in Cable Tunnel Fires, Nuclear Engineering and Design 241, 5263–5274, 2011.

[5] Mangs J and Keski-Rahkonen O. Full Scale Fire Experiments on Electronic Cabinets. VTT Publications 186, VTT Technical Research Centre of Finland, Espoo, Finland, 1994.

Experiences from Shutdown Fire PRA at Forsmark NPP

Erik Cederhorn, Maria Frisk
Risk Pilot AB, Sweden

Abstract

The cold shutdown mode has earlier been considered as a safe mode without a significant risk for a major accident. However during the last few decades knowledge has improved regarding risks during shutdown mode. Many activities are on-going during this period and the risk of fire occurrence may be affected. Due to an increased number of plant activities the integrity of the fire compartments may not be intact and this could lead to more extensive fire spreading. At the same time important barriers may be unavailable due to maintenance and a fire event could become critical. Time available for recoveries before fuel is exposed in the reactor pressure vessel after an initiating event i.e. fire event, which results in loss of residual heat removal, is in many cases significantly longer than 24 hours.

In order to increase realism in the analysis dependencies between plant risk and maintenance activities, i.e. different combinations of safety system alignments, during the shutdown period have been studied in detail. This has had an impact on the estimation of both fire ignition frequencies and probabilities for fire spreading between different compartments.

This paper will discuss the methodology applied to the fire PRA at Forsmark NPP during the cold shutdown period, with focus on fire frequency analysis and fire scenario analysis. The developing and implementation of fire analysis in the PRA and lessons learned from this will also be addressed

1. Introduction

This paper will discuss the methodology applied to the fire Probabilistic Risk Assessment (PRA) at Forsmark NPP during the cold shutdown period, with focus on fire frequency analysis and fire scenario analysis. The developing and implementation of fire analysis in the PRA and lessons learned from this will also be addressed.

Knowledge about risks during shut down period has improved during the last decades. Many plant activities are ongoing during this period and the risk of fire occurrence may be affected. Due to the increased number of plant activities the integrity of fire compartments may not be intact and this could lead to a more extensive fire spreading. At the same time important barriers may be unavailable due to maintenance and a fire event could become critical.

Efforts have been made to increase realism in the analyses by using improved methods. In order to increase realism in the analyses dependencies between plant risk and maintenance activities, i.e. different combinations of safety system alignments, during the shutdown period has been studied in detail. This has had an impact on both fire ignition frequencies and probabilities for fire spreading between different compartments.

The conditions during the shutdown period at Forsmark NPP must be known in order to enable the analysis. In order to perform the analysis it is relevant to know for example conditions during the different phases of the cold shutdown and also what initiating event that should be analysed.

1.1 Initiating event

The definition of an initiating event, at Forsmark NPP, is a disturbance in the nuclear power plant that requires one or more automatic or human initiated actions to bring the nuclear plant to a "safe" and "stable" mode. Loss of manual or automatic action can cause risk of a continuing process that may lead to release of radioactive materials to the environment.

During the cold shut down mode the initiating event is defined as an event that causes loss of residual heat removal. The different ways in which residual heat removal is maintained during the different phases is described in the section below.

1.2 Phases during cold shut down mode

At Forsmark NPP the analysis has been divided into six phases. The systems operating for residual heat removal differs depending on phase and this affects how a fire event should be analysed. In phase 1-3 the residual heat removal in the reactor pressure vessel, RPV, is performed by residual heat removal system (321).

These are the different phases during the cold shutdown mode:

- In phase 1 the reactor lid is mounted but the filling of the RPV has not started.
- In phase 2 the reactor lid is still mounted and the process for filling the RPV has started. At the end of this phase the reactor lid has been dismounted.
- In phase 3 the reactor lid is dismounted and the filling process of the pool above the RPV has begun.
- In phase 4 the pools in the reactor service room are filled with water. In this phase it is assumed that the majority of all maintenance is ongoing. During this phase the residual heat removal is performed by the residual heat removal system (321) and the fuel pit cooling and cleaning system (324), combined.
- Phase 5 see phase 2, the only difference is that the draining process of the RPV has begun.
- Phase 6 see phase 1.

The safety systems are divided into four independent trains A-D. During cold shut down mode it is assumed that two trains is unavailable for maintenance. In Forsmark 1 and 2 it is only possible for the combination of A and C or B and D to be unavailable at the same time. In Forsmark 3 all combinations of two trains can be unavailable at the same time.

Table 1. **Phase classification during cold shutdown**

	Reactor lid	**Water Level in RPV / Reactor pool**	**Operational system for residual heat removal system**
Phase 1	Mounted	Normal	The residual heat removal system (321) is cooling RPV
Phase 2	Mounted	Top filling/water level above streamlines.	The residual heat removal system (321) is cooling RPV
Phase 3	Dismounted	Reactor pool is empty	The residual heat removal system (321) is cooling RPV
Phase 4	Dismounted	Reactor hall pools are met	The residual heat removal system (321) and the fuel pit cooling and cleaning system (324) is cooling RPV and reactor halls pool together.
Phase 5	Dismounted	Reactor pool is empty	The residual heat removal system (321) is cooling RPV
Phase 6	Mounted	Normal	The residual heat removal system (321) is cooling RPV

The environment in the containment is assumed not to be inert during the whole shut down period and fires inside the containment can therefore occur.

2. Method

This chapter aims to give a short introduction and background to the method chosen for the analysis at Forsmark NPP. The standard method for the cold shutdown period at Forsmark is retrieved from reference [1]. The method used in this analysis is partly based on reference [2]. But due to limited resources the method could not be used completely.

2.1 Identify critical equipment

Critical components are those components that directly or indirectly are included in the safety related systems and also are included in the PRA model. When the critical components have been identified the failure mode in case of fire needs to be determined. Only active equipment which need power supply is assumed to be affected by the fire. Passive objects such as a heat exchanger can be assumed to be unaffected by a fire.

2.2 Mapping of electrical system

Mapping of the electrical system is a very important step in the fire analysis. The electrical systems are built with circuits that form networks and branches with cables, loads, cabinets and breaking points. The electrical networks are widespread throughout the building and therefore sensitive to area events such as fire.

2.3 Identify fire events

Fire can be assumed to occur in all rooms in the power plant. Rooms containing objects included in the PRA model are analysed. Fire spreading from other rooms into these rooms must be taken into account.

2.4 Screening

A screening of fires in fire compartments or fire cells which implies an initiating event, the screening criterion, i.e. fires that causes loss of residual heat removal, should be done. A fire should be assumed to destroy all the electrical system in the fire compartment or fire cell. This approach can be sensitive to errors in electrical mapping which is fundamental to the screening process. This approach does not consider if a fire event causes a degraded barrier, only events that lead to an initiating event will be taken into account.

2.5 Data analysis

Frequency for fire occurrence should be calculated for each room included in the fire compartments or fire cells that have been screened out. The frequency depends on the type of room and is calculated from fire statistics from the Swedish and Finnish NPPs countries. The statistics are based on fires during shut down periods. In the statistics there is information about whether the fire occurred due to on-going work in the room. The risk for fire during on-going work could be much higher, in order of twice the risk.

For example assume that power supply train A is unavailable because of maintenance. This means that the all rooms that contain at least some equipment for train A the room gets a higher fire frequency, this is relevant in the PSA-model because the room could also contain equipment from train B. In rooms that only contain equipment from train B, which is not shut down for maintenance, it is assumed that no work is in progress and the lower frequency should be applied.

If the fire compartment or fire cell contains equipment from the train which is unavailable for maintenance the frequency for on-going work should be applied. So depending on what train combination that are unavailable for maintenance the frequency varies. The frequency should be allocated to the different phases in proportion to its length.

2.5.1 Fire extinguishing

Manual and automatic extinguishing is not modeled explicitly in the PRA model. However successful firefighting efforts could be considered when calculating fire occurrence frequencies.

2.5.2 Fire spreading

It is possible for the fire to spread inside the fire compartment or fire cell. A probability for the fire to spread to the next room can be applied.

2.5.3 Impact of manual operation

Fire could have an impact on manual operations, Post-incident actions (Category C), it could for example affect the place where local maneuvers are performed or the information at the main control room could be affected. Therefore the probability for failure of manual operations should be re-estimated during an on-going fire.

2.5.4 Detailed analysis

If needed a more detailed analysis can be done, focusing on analyzing the compartments that give high core damage frequency.

3. Analysis

In this chapter the fire analysis performed at Forsmark NNP is described. An extensive work of mapping of the electrical system, according to the described method, has been performed and applied in the full power PRA model. Figure 1, below, illustrates the entire analysis process. This chapter will explain and describe the different parts of the process.

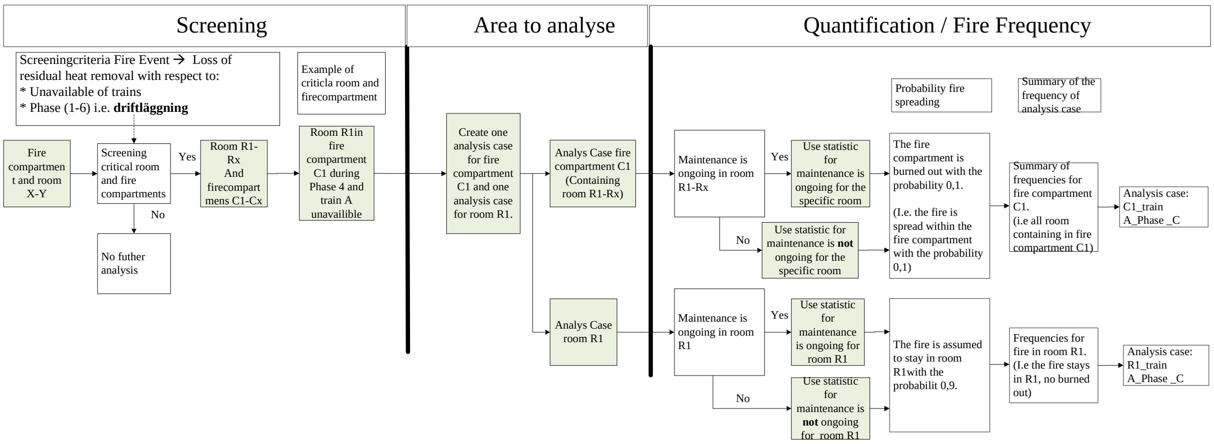


Figure 1. **Fire analysis process**

3.1 Screening process

A screening was done for fire in fire compartment and fire in room that leads to initiating event, i.e. loss off residual heat removal, see figure 2. To be considered a critical compartment a fire need to cause loss of residual heat removal if the whole compartment is burnt out and to be a critical room fire needs to cause loss of residual heat removal if the whole room is burnt out. One critical compartment could consist of several critical rooms.

Since there are different conditions, depending on phase during the cold shutdown and what train combination is unavailable, the screening was done for each train combination unavailable during every phase. When a list of fire compartments and rooms needed to be analysed further was done the analysis cases was created and described in the next section.

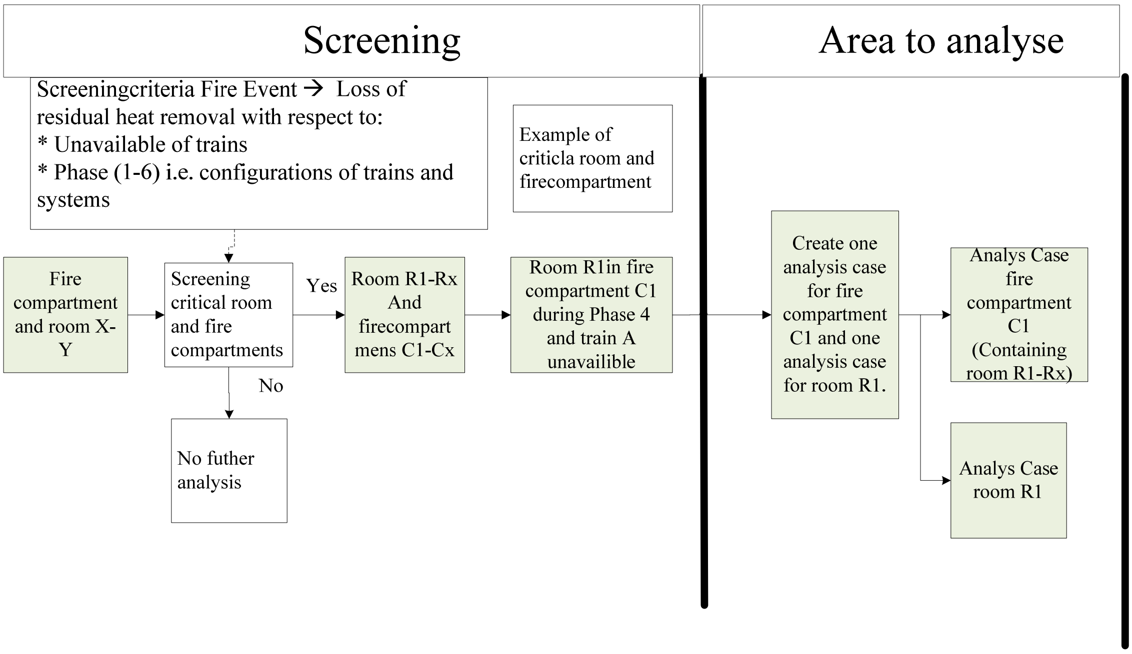


Figure 2. **Screening process**

3.2 Analysis cases and frequency for fire occurrence

For each critical room or compartment analysis cases was created for the critical conditions i.e. phase and unavailable train combinations.

The frequency for fire occurrence in each room or fire compartment is calculated depending on several parameters described in the method. The process is described in figure 3.

- Type of room
- On-going work/no work on-going
- Length of the phase

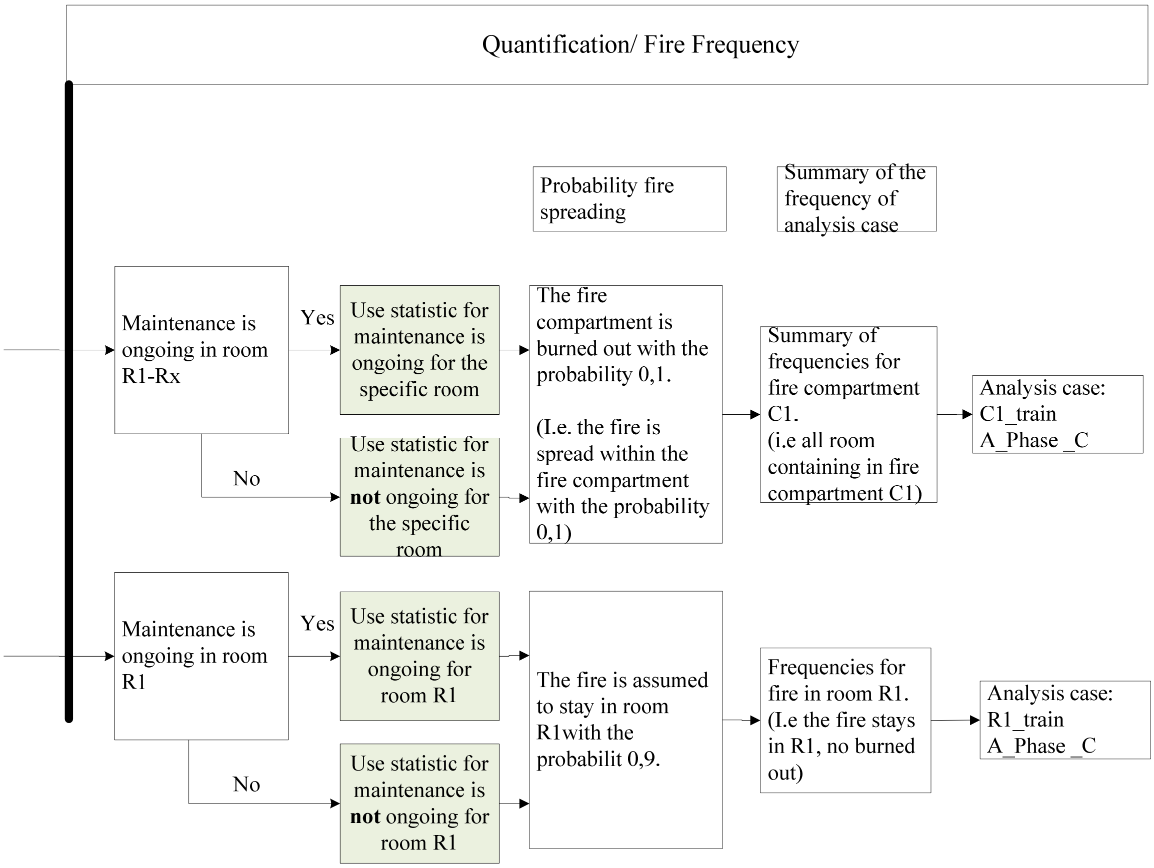


Figure 3. **Quantification/Fire frequency**

3.2.1 Fire spreading

Fire in fire compartment was assumed to burn out the whole compartment with a probability of 0.1 and with a probability of 0.9 the fire was assumed to only burn out the room.

3.3 Walk downs

Walk downs have been performed with purpose to find deficiencies in the mapping of fire compartments. The main reason for the deficiencies in the mapping of compartment is that the layout in some cases does not correspond to reality.

4. Results

In order to increase realism, dependencies between plant risk and maintenance activities, i.e. different combinations of safety system alignments, during the shutdown mode have been studied in detail. This has had an impact on both fire occurrence frequencies.

Fire during the cold shutdown period leading to loss off residual heat removal gives a core damage frequency in the magnitude of 1E-6 and is 4,5% of the total core damage frequency for Forsmark 3. Fire during power operation gives a core damage frequency in the magnitude of 1E-7 and is 3% of the total core damage frequency for Forsmark 3. Fire is not a dominating initiating event. But the risk of getting a critical fire is greater during cold shutdown mode than during power operation.

The risk for fire during cold shutdown period is much higher that during power operation. The reason for that is the risk increase because of on-going maintenance.

5. Conclusion

The result of this analysis shows that the most critical phases with respect to a fire event are the phases when upper head (reactor vessel lid) still is mounted. If a fire event occurs during any of these phases it leads to loss of residual heat removal and the time of recovery is relatively short.

Due to an increased number of plant activities during the cold shutdown period the integrity of fire compartments may not be intact. This could lead to an even more extensive fire spreading. At the same time important barriers may be unavailable due to maintenance and a fire event could become critical. The risk for a fire to occur and to be critical is more probable during cold shutdown than during power operation. Therefore it is very important to analyse and to implement this in the PRA studies. On the other hand, available time for recoveries before fuel is uncovered in the reactor pressure vessel after an initiating event, i.e. fire event that result in loss of residual heat removal, is in many cases significantly longer than 24 hours. The reason for this is that during the period (phase 4) when most of the maintenance is ongoing all pools are filled with water and a large water volume must be boiled off before fuel is uncovered. This means that other aspects related to fire events during the cold shutdown period might be more relevant. For example consequences originating from the spent fuel pool after a fire event or combinations of fire event that could cause a leakage from RPV. The results shows that the only period when a fire event could lead to exposed fuel within 24 hours are during phase 1, 2 and 3.

Another conclusion is that the frequency of fire occurrence differs significantly between the rooms where maintenance is on-going and rooms where maintenance is not on-going. On the other hand the probability of successfully fire extinguishing is significantly higher in rooms where work is in progress. Fire frequencies are generally higher during the cold shutdown mode compared to operating mode.

During our work with fire analysis the points below with possible further development were found.

- More detailed analysis of how human error is affected by the fire
- Secondary events caused by fire leading to combined events, for example LOCA caused by fire.
- Secondary fire events caused by the initiating event.
- Fire when all fuel is unloaded from reactor pressure vessel and put into spent fuel pool.
- A more detailed analysis of how human error is affected by a fire

References

[1] Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes, IAEA, Vienna, 2000 IAEA-TECDOC-1144, ISSN 1011-4289

[2] Fire PRA Methodology for Nuclear Power Facilities, EPRI/NRC-RES, vol 2 detailed methodology, EPRI 1011989, NUREG/CR-6850

Latest Extensions of Loviisa Fire PRA

Antti Paajanen and Sami Sirén
Fortum Power and Heat Oy, Finland

Abstract

The level 1 fire PRA for power operation mode of Loviisa NPP was completed in 1997 and has been annually updated since then. In recent years the fire PRA has been extended to cover power operation also for level 2 and shutdown states for levels 1 and 2.

The level 2 fire PRA models for power operation mode and shutdown states of Loviisa NPP were completed in 2012 and 2013, respectively. The level 2 fire PRA utilizes existing level 1 fire PRA fault-trees and level 2 internal event PRA event trees. As the Level 2 fire PRA model was built on top of the existing level 1 fire PRA model, which was not developed with the level 2 PRA in mind, and therefore e.g. did not use exchange events, there were some limitations and challenges in the way level 2 fire PRA analysis cases could be implemented. As a result, the level 2 fire PRA is calculated using an external utility program which goes through two separate event tree loop-structures and inputs correct parameters (analysis case, initiating event and core melt bin) into the risk analysis software using command line parameters and SQL programming language queries. The first loop determines the core melt bin and the accident progression category and the second loop determines the initiating fire event (room and the type of the initiating event).

1. Introduction

Level 1 fire PRA has been part of Loviisa NPP PRA work since the late 1980s. The power operation fire PRA was completed in 1997 and revealed safety issues that were answered by several plant modifications, such as the backup Residual Heat Removal (RHR) System. Since then, the analysis has been annually updated.

As the internal events PRA models for shutdown states and level 2 were developed, also (the extension of) the fire PRA to include these became possible. The main motivation for these extensions is to complete the scope of both level 1 and 2 PRA to cover all internal, severe weather, flooding, seismic and fire events for all plant operating modes.

The level 1 shutdown fire PRA relies heavily on the internal events PRA for shutdown states, as well as the power operation fire PRA. As the level 2 is built on top of the level 1 model, the assumptions and technical solutions of level 1 model guide the development of the level 2 fire PRA.

The extension of the level 1 fire PRA to shutdown states is presented in Chapter 2. The methodology and challenges faced in the further extension to level 2 are presented in Chapter 3. The results and conclusions are presented in Chapters 4 and 5.

2. Level 1 Fire PRA

As the level 1 power operation fire PRA has been reported earlier [1], the description in Chapter 2.1 is restricted to what is needed for the comprehension of the further chapters.

2.1 Power operation

The fire PRA methodology developed for Loviisa NPP included

- an identification of initiating events (IEs) based on the internal events analysis,
- an extensive investigation of cable routings of the equipment related to the identified IEs and safety equipment,
- an identification of fire rooms using IE fault trees,
- a fire frequency estimation using both plant data and international fire event data,
- a two-phase fire scenario development, and
- an estimation of conditional core damage frequency based on the IEs and safety equipment failures.

In the first phase of the scenario development, very conservative assumptions about the fire damage potential were used, assuming all equipment and cabling in the ignition room to fail. In the second phase, the most significant fire scenarios were studied in more detail to remove excess conservatism. The room-based fire frequency was allocated to separate ignition sources, various fire simulation codes were used to analyse the fire damage potential and fire suppression possibilities were taken into account also in the ignition room. Less significant scenarios were not studied further in the second phase. The current level 1 fire PRA consists of more than 700 fire scenarios.

2.2 Shutdown

The shutdown fire PRA was performed using the same procedure as the power operation fire PRA but applying it to shutdown states. As the shutdown PRA is divided into 15 plant operating states (POSs), each identified fire-induced IE was only modelled in some sub-set of POSs. Furthermore, the plant configuration and success criteria for the relevant IEs during the hot shutdown and start-up states were considered similar to power operation, so applicable power operation fire PRA scenarios were used. New fire scenarios were only needed for the cold POSs, greatly reducing the work load.

Due to the past investigations and simplifying assumptions (e.g. motor operated valves can be manually operated during shutdown states in case of power cable failures), most of the required cable routing information was already available for the level 1 shutdown fire PRA.

The fire frequency for each room type was estimated using an empirical Bayes method [2] and the same databases as in power operation fire PRA, but only considering fire events during shutdown. For the hot POSs the data was insufficient (zero events before refuelling and six after refuelling) so power operation fire frequencies were used instead. For the start-up POSs the all power operation fire frequencies were multiplied by 3.5 to account for the difference in the estimated overall frequency. For the cold POSs, there was sufficient data for a relatively reliable estimation of fire frequencies for all room types. The frequencies were allocated to individual ignition rooms using the same methods and parameters as in power operation fire PRA.

The important fire areas are largely the same during the shutdown states and power operation. Therefore the detailed analyses performed in the two-phase fire scenario development were already available and easily implemented.

The remaining conservatism in the shutdown fire scenarios was mainly related to the maintenance activities. The number of available safety trains after the fire can in some cases vary from 0 to 2, depending on whether the maintenance unavailability is modelled as best-estimate or according to the minimum requirements in Technical Specifications as was done in the internal event PRA. In the internal event PRA this was not such a big issue, because an IE causes rarely also the loss of two safety trains, something that is totally possible due to fire damage. For the shutdown fire PRA, it was essential to choose the best-estimate perspective to get realistic results, as most of the time at least 3 out of 4 safety trains are available even during cold shutdown.

3. Level 2 Fire PRA

The implementation of level 2 fire PRA model for Loviisa NPP is similar to other level 2 models (internal events, flooding and severe weather events). As level 1 fire PRA for Loviisa NPP has been developed using fault trees, a level 1 interface event tree has been developed for each initiating event. Using these interface trees it is possible to quantify the frequency of each core melt bin (CMB) depending on the successes and failures of the safety systems. These CMBs function as initiating events for the level 2 containment events trees. As the interface trees and containment event trees are linked, the boundary condition sets (BC sets) and successes and failures from level 1 are automatically taken into account in the level 2 analyses.

The interface event trees for the fire-induced initiating events differ from the other interface trees as it must also be possible to alter the fire room for each initiating event, in addition to altering the initiating CMB of the containment event trees. Because of the limitations of the risk analysis software, these alterations must be done for each initiating event and fire room either by manual changes in the risk analysis software or by SQL queries using an external utility program.

This level 2 fire PRA approach was selected because any other approach would have required significant changes to the structure of the level 1 fault trees.

3.1 Input data

In order to assess the impact of fire-induced initiating events on the severe accident management (SAM) systems in every fire room, an extensive investigation of cable routings of the equipment related to the identified IEs and safety equipment was conducted. This data was mainly retrieved from plant documentation and it was supplemented with details, such as unmapped cable routings and locations of the SAM equipment, gathered during several visits to the plant.

The level 2 fire PRA model assumes that if there is a fire in a room in which there is a cable controlling or powering a SAM system or a part of the SAM system itself, that SAM system is lost. This information was transferred into the level 2 fire PRA model using fire room specific boundary condition sets. The fire frequencies for each fire room have been estimated during the development of level 1 fire PRA and these frequencies were not changed during the development of level 2 fire PRA.

3.2 Plant operating states, initiating events and core melt bins

The fire PRA for Loviisa NPP recognizes 10 different fire-induced initiating events for power operation and 14 initiating events for shutdown plant operating states with different sets of lost safety related equipment. As mentioned in Chapter 2.2, the shutdown PRA is divided into 15 different POSs. For example, power operation is divided into 3 different POSs, hot shutdown into

6 POSs, cold shutdown into 4 POSs and refuelling into 3 POSs. For the sake of simplicity, in this chapter shutdown POSs are divided into cold shutdown, refuelling and hot shutdown. The initiating events for each plant operating state are presented in Table 1.

Table 1. **Fire-induced initiating events recognized in the level 2 fire PRA**

		Plant Operating States			
Initiating event	**ID**	**Power operation**	**Hot shutdown**	**Cold shutdown**	**Refueling**
Loss of DC power	LDCP	X	X	X	X
Loss of instrumentation room ventilation	LIRV	X	X	X	X
Loss of main feed water	LMFW	X	X		
Loss of offsite power	LOOP	X	X	X	X
Medium loss of coolant accident	MLOCA	X	X		
Pressurizer loss of coolant accident	PLOCA	X	X		
Reactor trip	RT	X			
Small loss of coolant accident	SLOCA	X	X		
Total loss of feed water	TLFW	X	X		
Total loss of service water	TLSW	X	X	X	X
Total loss of residual heat removal	TLRR			X	X
Partial loss of residual heat removal	PLRR			X	X
Partial loss of service water	PLSW			X	X
Generic core damage during cold shutdown and refueling	CD			X	X

These initiating events have also been analyzed in the level 2 PRA as internal initiating events, where the initiators are, for example, component failures instead of fires.

Fire-induced initiating events have one or more variants, which cause varying levels of failures of safety related equipment to the plant. These failures are modeled using different boundary condition sets for each variant. Which variant is used in which fire room depends on the equipment and cablings located in the particular room. Some of the variants are estimated to cause so significant damage to the plant safety systems that they are estimated to result in a core damage accident (CD in the listing above), regardless of safety systems.

The level 2 fire PRA recognizes 86 different core melt bins: 56 for power operation, 18 for cold shutdown (including refueling) and 12 for hot shutdown. These CMBs are divided into 3 categories: containment bins (N-bins) where the containment building remains operational after the core melt, containment bypass bins (B-bins) where the containment building is bypassed (e.g. large steam generator tube rupture) and X-bins where the status of the containment building is difficult to estimate using PRA methods (e.g. drop of heavy load during shutdown).

3.3 Level 2 Fire PRA event trees

As mentioned in the beginning of Chapter 3, the level 2 fire PRA model consists of interface event trees and containment event trees.

The level 2 fire PRA model has 24 interface event trees. To simplify the level 2 PRA model, power operation interface event trees are used to model the hot shutdown initiating events and all cold shutdown initiating events are modeled using a single set of cold shutdown interface event trees. An example of an interface event tree can be seen in Figure 1.

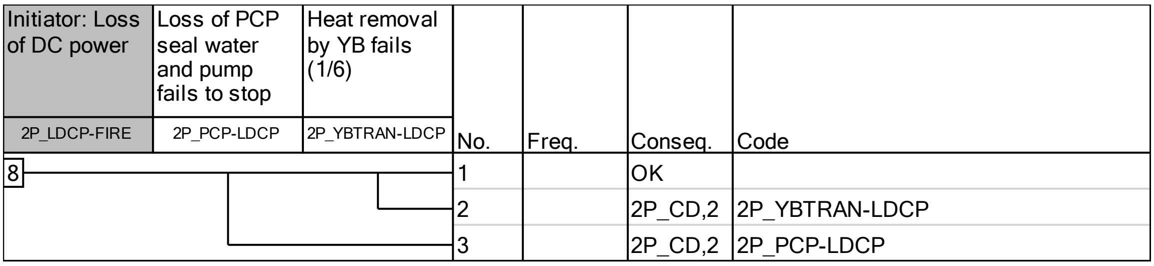


Figure 1. **Interface event tree of Loss of DC Power initiating event**

The fire room can be selected by altering the initiating event branch-point number (represented by a boxed 8 in Figure 1.) which points to a specific fire room gate in the level 1 Loss of DC Power fault tree and also includes a specific boundary condition set. The links to the containment event tree can be seen in the column 'Conseq.' where the initiating event results either in a success or a CMB according to the results of the top events.

There are two containment event trees in the level 2 fire PRA model: one for power operation and one for cold shutdown states. The containment event tree for power operation events can be seen in Figure 2.

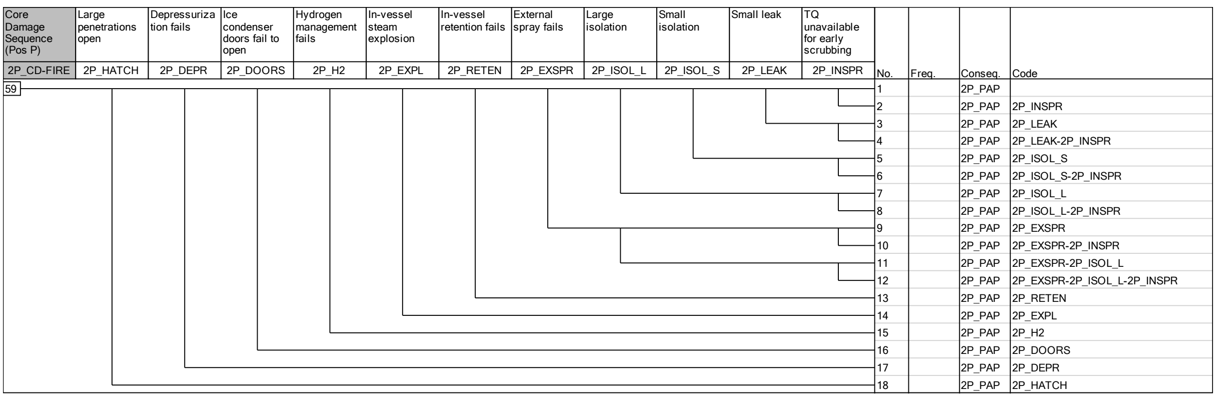


Figure 2. **The containment event tree for power operation**

The CMB, which functions as the initiating event, can be selected by altering the initiating event branch-point number (represented by a boxed 59 in Figure 2.) which points to a specific consequence in the interface event tree and also includes a specific boundary condition set. In the containment event tree, the initiating event frequency is divided into 18 different accident progression categories (APC in the 'Conseq.' column) according to the results of the top events.

In case of a containment bypass CMB, it is assumed that water in the containment building is lost due to the containment bypass and there is no water to flood the reactor cavity. This leads to the failure of the in-vessel retention of corium (top event 2P_RETEN in Figure 2) and all of the remaining initiating frequency is designated to APC13. In a case of a X-CMB, the containment event tree is not used and a pre-determined fraction, e.g. 50 % of the core damage frequency is assumed to result in a large radioactive release.

3.4 Boundary conditions sets

The boundary condition sets are lists of conditions, which affect fault trees and event trees of PRA models by changing statuses of basic events, house events or gates. As presented in Chapter 3.3, these configurations are inherited from interface event trees to containment event trees through links.

In the level 2 fire PRA, every plant operating state, fire room, initiating event and core melt bin has a particular boundary condition set, which are used in conjunction in order to individually alter the fault trees and event trees for each analysis case. As a result, this eliminates the need for excessive interface event trees and containment event trees, which could potentially rapidly inflate the PRA model.

The level 2 fire PRA boundary condition sets use the following designations:

- Fire room boundary condition sets are designated as *2[POS]_[Fire Room]/[Initiating Event]-[Initiating Event Variant]*. E.g. *2P_V0830/LDCP-01.*This boundary condition set includes boundary conditions of the power operating state, fire room V0830 and Loss of DC power variant 01.
- CMB boundary condition sets are designated as *2[POS]_P[CMB]-[Initiating Event]*. E.g. *2P_PXLD-LDCP*. This boundary condition set includes boundary conditions of the power operating state, core melt bin XLD and Loss of DC power.

3.5 Calculation routine

The level 2 fire PRA consists of a large amount of analysis cases, which all require individual initiation procedures, as presented in Chapter 3. It is possible to carry out these initiations manually using the risk analysis software user interface but the amount of time required makes this approach prohibitive. Therefore Fortum has developed an external utility program which goes through two separate event tree loop-structures and inputs correct parameters (analysis case, initiating event branch-point number and core melt bin branch-point number) into the risk analysis software using command line parameters and SQL queries.

A simplified calculation routine of a single initiating event is presented below.

At first, the containment event tree initiating event branch-point number is set to the first core melt bin alternative, as pictured in Figure 3. This is done by a SQL query, which will determine the correct event tree in the database on the basis of the name of the analysis case listed in the core melt bin loop.

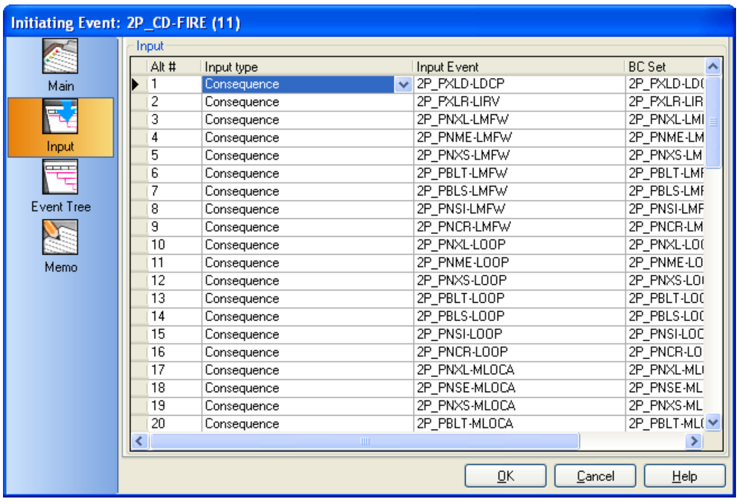


Figure 3. **List of alternative initiating events in a containment event tree**

The interface event tree initiating event branch-point number is set to the first fire room alternative, as pictured in Figure 4. As in the core melt loop, this is done similarly by a SQL query.

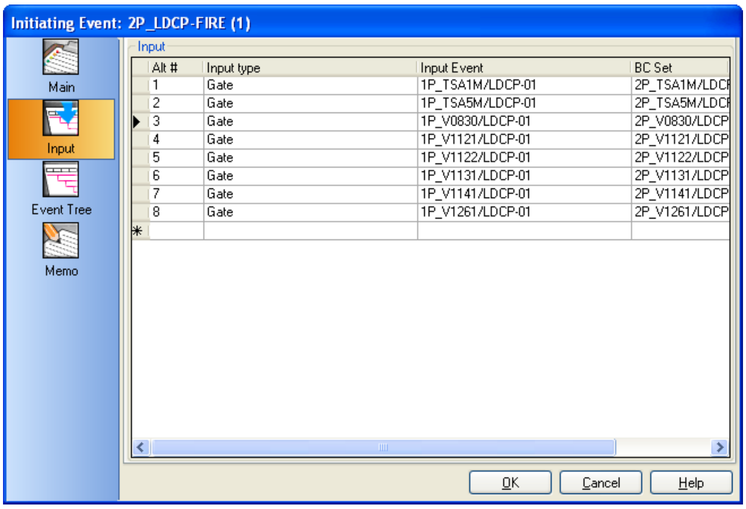


Figure 4. **List of alternative initiating events in an interface event tree**

When the fire room has been set, the utility program will initiate the calculation of the level 2 analysis case defined by both the core melt bin loop and the fire room loop using a command line command.

When the analysis case has been calculated, the fire room loop will move to the next listed initiating fire room. If there are no more fire rooms left in the loop list, the fire room loop will close and the core melt bin loop will move to the next listed initiating core melt bin. If there are no more core melt bins left in the loop list, the core melt bin loop will close as well.

When both loops have closed, the minimum cut sets and importance analyses are calculated using a separate command line command.

This is repeated for every initiating event and every plant operating state until all required analysis cases have been analyzed.

The current level 2 fire PRA model consists of over 12000 analysis cases. If a single modern workstation is used, it takes approximately 6 days 10 hours to calculate these cases as there is a slight waiting time before and after each analysis case. Consequently, the external utility program is designed in a way that allows the user to calculate only a certain part of the analysis cases, e.g. power operation or a particular initiating event, allowing multiple users to split the analysis cases and thus reduce the calculation time.

4. Results

The main results of the Loviisa PRA (both level 1 and level 2) are presented in Figure 5. The annual (power operation and shutdown) core damage frequency (CDF) caused by fire-induced initiating events is 6,9E-6/a, which is 30 % of the total core damage frequency. The large release frequency (LRF) caused by fire-induced initiating events is 7,1E-7/a, which is 9 % of the total annual core damage frequency. The LRF/CDF ratio of fire-induced events is 10,2 %, which means that in most fire events, the SAM systems are not damaged by the fires.

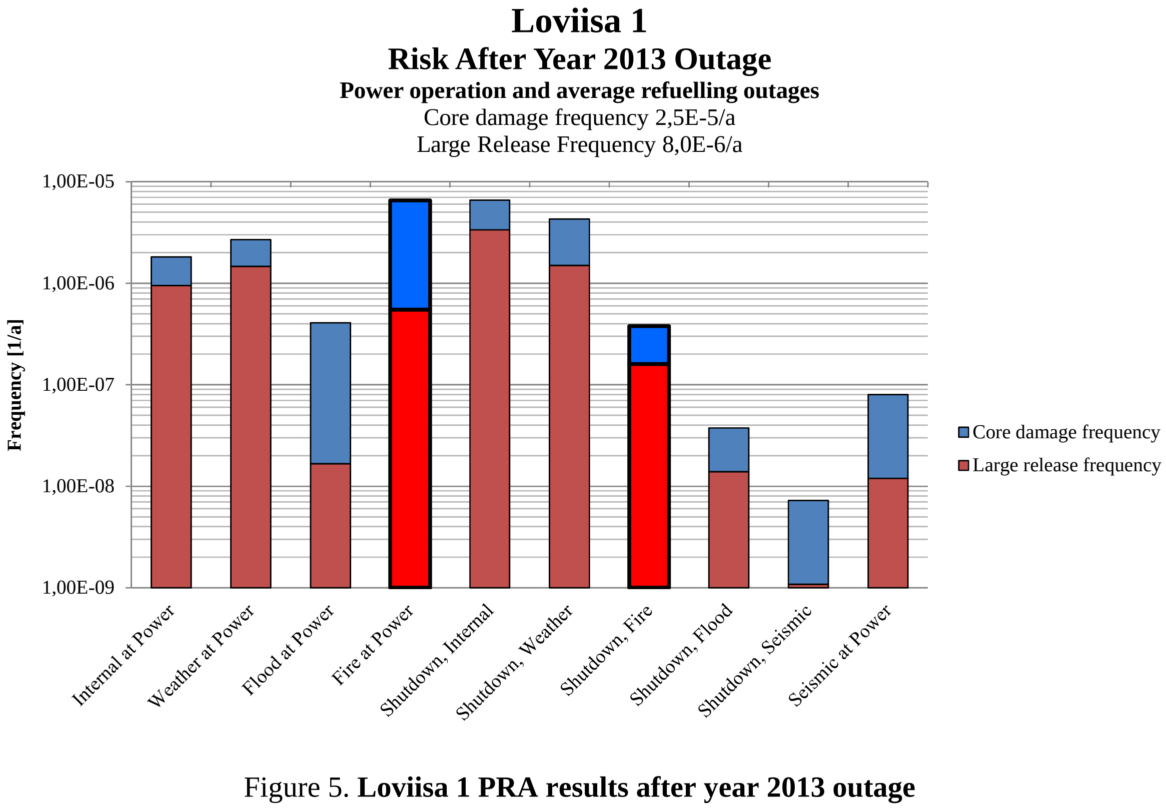


Figure 5. **Loviisa 1 PRA results after year 2013 outage**

The main results of the fire PRA (both level 1 and 2) are presented in Figure 6.

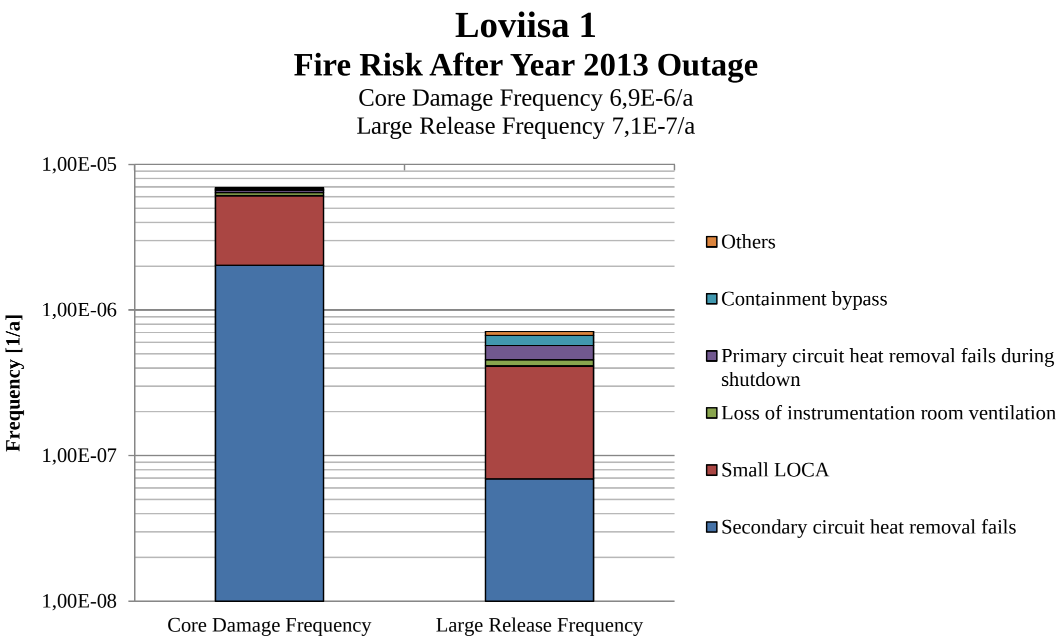


Figure 6. **Fire PRA results after year 2013 outage**

In addition, the results include measures of importance for every basic event therefore making it possible to sort all fire rooms according to their contribution to the overall fire risk. This will be helpful in numerous applications, e.g. in improving the plant fire brigade efficiency.

5. Conclusions

Although the structurally large level 1 fire PRA model gave rise to special challenges in developing the level 2 fire PRA, ultimately it was possible to create a complete and robust level 2 fire PRA in a relatively short time frame with the help of external utility programs.

In the level 1 fire PRA, the most important fire rooms, usually cable corridors, are located in the vicinity of the main control building and emergency diesel generator area. As the SAM systems are mainly located in the reactor building, where the fire frequencies are relatively low, and are powered by a separate SAM diesel system isolated from other diesel generators, only a minor portion of the fire-induced initiating events result in a failure of the containment and a large radioactive release.

References

[1] M. Lehto et al. (1996), Fire risk analysis for Loviisa 1 during power operation, Proc. PSA'96, Park City, Utah, Sept. 29-Oct. 3, 1996.

[2] J. K. Vaurio and K. E. Jänkälä (2006), Evaluation and comparison of estimation methods for failure rates and probabilities, Reliability Engineering and Systems Safety, 91, pp. 209-221, 2006.

Conducting Fire PSA for the Post-commercial Shutdown Phase

Siegfried Babst[a], Marina Röwekamp[b], Michael Schwarz[c], Michael Türschmann[a]
[a] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Berlin, Germany
[b] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, Germany
[c] EnBW Kernkraft GmbH Kernkraftwerk Philippsburg, Germany

Abstract

The German nuclear power plant (NPP) Philippsburg, Unit 1 has stopped commercial operation in March 2011 due to the decision of the German government after the reactor accidents of Fukushima Dai-ichi to decommission the eight oldest of the in total seventeen NPP units in Germany. Since then this boiling water reactor (BWR 69) type plant built to earlier standards is in a post-commercial plant operational shutdown state. During the post-commercial operational phase before decommissioning the reactor pressure vessel of this plant does no longer contain the fuel elements, all of which have been removed to the spent fuel pool (SFP).

The licensee has recently asked the regulatory body for the permit of a technical plant modification concerning the spent fuel pool cooling. Major differences of the intended plant modifications compared with to the original situation are the number of emergency power supply systems available and the systems used for cooling the spent fuel. According to the most recent German regulatory requirements, any plant modification with the potential to change core damage, fuel damage or large release frequencies has to be probabilistically assessed in addition to the deterministic assessment.

In order to quantitatively compare the original spent fuel cooling and the intended modified one, a PSA for the post-commercial operation shutdown phase has been performed. These probabilistic analyses also covered the plant internal hazard fire.

The technical document on PSA methods supplementing the German PSA Guide outlines in detail the approach for Fire PSA for full power operational states. For the Philippsburg NPP, Unit 1, a Fire PSA according to the state-of-the-art requirements provided in this guidance document is available. The comparative probabilistic fire analyses conducted for the two alternatives of spent fuel pool cooling in the post-commercial shutdown phase are principally based on the information and data provided for Level 1 full power operation Fire PSA. It has been demonstrated, if and to what extent these data have to be modified for modelling the fire specific aspects of the post-commercial operation shutdown phase as realistically as possible.

For carrying out a comparison of the Fire PSA for the above mentioned two alternatives of spent fuel pool cooling in this phase, the already existing PSA plant model has been systematically and partly already automatically extended considering so-called fire equipment and fire propagation lists. The Fire PSA was conducted applying a compartment-wise screening approach. For each compartment identified within a conservative fire specific building partitioning, the annual frequency of fuel damage states has estimated.

For each relevant component the fire equipment list contains its location (compartment), and the routing of the corresponding power supply and instrumentation and control (I&C) cables. The fire propagation list contains the information on the adjacent compartments to each fire compartment and the corresponding fire propagation probabilities between compartments directly adjacent to each other.

The paper presents the approach for conducting the Fire PSA for the two alternatives and discusses their results. Based on the application example, methodological conclusions are drawn for performing Fire PSA for low power and shutdown states including the post-commercial shutdown operation phase.

1. Introduction

The German nuclear power plant (NPP) Philippsburg, Unit 1 has stopped commercial operation in March 2011 due to the decision of the German government after the reactor accidents of Fukushima Dai-ichi to decommission the eight oldest of the in total seventeen NPP units in Germany. Since then this boiling water reactor (BWR 69) type plant built to earlier standards is in a longer duration post-commercial shutdown state. During the post-commercial operational phase before decommissioning the reactor pressure vessel of this plant does no longer contain the fuel elements, which entirely are stored in the spent fuel pool (SFP).

The licensee intends to apply for a technical plant modification with respect to the SFP cooling which needs to be licensed by the regulatory body. Major differences of the intended plant modification in comparison to the original situation are the number of available emergency power supply systems and the systems used for cooling of the spent fuel pool. According to the actual German regulatory requirements any plant modification with the potential to change core damage, fuel damage or large release frequencies has to be probabilistically assessed in addition to the deterministic assessment.

In order to be able to quantitatively compare the original spent fuel pool cooling and the modified cooling as intended, a probabilistic analysis has been performed for the post-commercial operation shutdown phase.

Between GRS and the licensee of the Philippsburg NPP, there has been a long-standing fruitful working relationship with the Unit 1 been taken as BWR-69 reference plant for several research activities in the field of PSA. A variety of research and development activities carried out by GRS could be validated using plant specific data and the operating experience of KKP 1. As an example, a complete state-of-the art Level 1 Fire PSA was carried by GRS on behalf of the German Federal Office for Radiation Protection (BfS) as well as further research and development studies with regard to enhancements in Fire PSA based on this reference plant [1]. This resulted in a strong interest of BfS as well as of the licensee to ask GRS for performing probabilistic fire analyses for the post-commercial shutdown phase. These activities represent a pilot project, on the one hand due to the fact that there is up to the time being no guidance available for low power and shutdown Fire PSA and, on the other hand because of questions arising from a comparative analysis. However, one non-negligible advantage of this project was that the Fire PSA activities could be used to incorporate internal hazards more generally in the approach for performing site specific Hazards PSA (see [2]).

2. Conducting Fire PSA

In order to carry out a probabilistic analysis of the effects of fires in nuclear power plants on risk, the layout of the plant is sub-divided into spatial units. This procedure is called plant partitioning.

These spatial units are referred to as compartments. As a general rule, this partitioning performed for the purpose of the analysis is carried out by using the existing structure of rooms or plant areas for which a nomenclature already exists. Depending on the necessary degree of detail of the analysis, a finer or coarser spatial division may be chosen. The fire induced risk of the NPP is the sum of the fire induced risks posed by the individual compartments of the partitioning. Here, it is assumed that the entire compartments describe at least all relevant buildings of the NPP and that there is no overlap of any pairs of compartments. Buildings are referred to as relevant ones, if in the event of a fire any equipment inside the building might be damaged whose failure would contribute to the target value of the analysis. The modelling with event and fault trees depends on the analysis target of the study. In the case of a Level 1 full power operation PSA, the target is the determination of the core damage frequency. For comparing risk, reliability and effectivity of different fuel pool cooling systems presented here, it is the estimation of the frequency of fuel damage states. In this context, the fuel damage state is characterised by the fact that the fuel assemblies in the spent fuel pool are no longer covered by coolant, so that fuel rod integrity is impaired by uncontrolled heat-up.

In order to determine the risk posed by a fire inside a compartment applying a quantifiable plant model in line with the analysis target, the following data and information are required:

- Compartment specific fire occurrence frequency;
- Equipment lists for all compartments including cables,
- Equipment classification with respect to their potential risk significance:
  - (1) Items important to safety (so-called PSA equipment),
  - (2) Items, which in case of their fire induced failure may contribute to an initiating event (so-called IE equipment), and
  - (3) Other equipment (irrelevant for fire risk analysis);
- Arrangement of compartments in the building (neighbouring influences) and probabilities of fire spreading from a compartment to an adjacent one;
- Compartment related fire damage probabilities for all class (1) and (2) equipment.

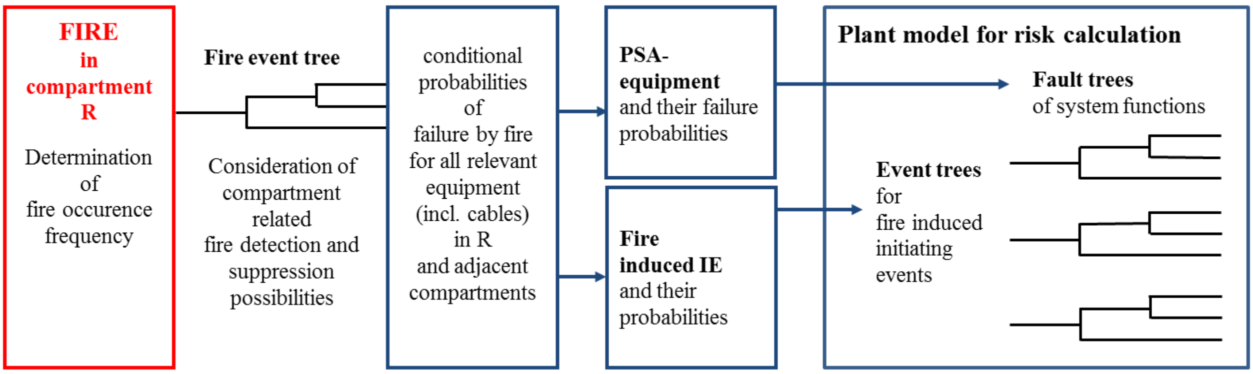


Figure 1. **Calculation of risk as consequence of fire events in a given compartment R**

The compartment related fire damage probabilities as well as the probabilities of fire spreading are determined by means of fire event trees, considering available information and knowledge about fire detection and alarm and fire extinguishing as well as about possible operator actions. It is assumed that the conditional fire induced damage probability is the same for the entire equipment installed in a given compartment. This probability is also referred to as the compartment damage probability.

By means of this (here only roughly described) method (see also figure 1), Level 1 Fire PSA for full power operation has been performed for KKP 1 [1] a few years ago, which also means that the partitioning of the buildings into different compartments and all necessary compartment related data and information are now available for use and application in further analyses.

The technical document on PSA methods [3] supplementing the German PSA Guide outlines explicitly only an approach for conducting Fire PSA for full power plant operation. In the recent draft of a planned state-of-the-art supplementary volume to the Guideline [4], it is stated that the methodology presented may also be applied to low-power and shutdown plant operational states. Differences do exist in the data and information needed for analysis; these have to be considered when applying the approach.

For performing a Fire PSA for low power and shutdown states, it is assumed here that

- a comprehensive Fire PSA for full power operation as well as
- a comprehensive PSA for low power and shutdown plant operational states including the longer duration post-commercial operational phase

are available and that reference can be made to the models and data used in these analyses.

The plant partitioning into spatial units as described above may largely be adopted from the Fire PSA for full power operational states. The division into different plant operation states may be adopted from low power and shutdown PSA, see also remarks in Chapter 4.

In accordance with the procedure for full power operation, the fire induced risk has to be calculated for each compartment and each plant operational state. For the modelling, the risk measure has to be correspondingly determined in advance.

For calculating of the compartment related fire risk, the following modifications and extensions have to be considered with respect to the input data and information:

- State with regard to equipment installed inside a compartment:
  The equipment installed in each compartment has have to be re-classified according to their safety related function for each phase of low power and shutdown plant operational states with respect to their risk significance.
- State with regard to fire suppression in each compartment:
  Depending on the plant operational state, modifications with respect to the status of fire suppression are possible, e.g. changes in the status of fire barriers, in the amount and distribution of fire loads, consideration of the effects of maintenance and repair work, changes in the number of humans present in the compartment and the duration of presence.
- Compartment related initiating events:
  In case of a fire induced failure of all equipment present in a room, the list of the potential initiating events has to be checked subject to the plant operation state.

3. Intended Modification of the Spent Fuel Pool Cooling

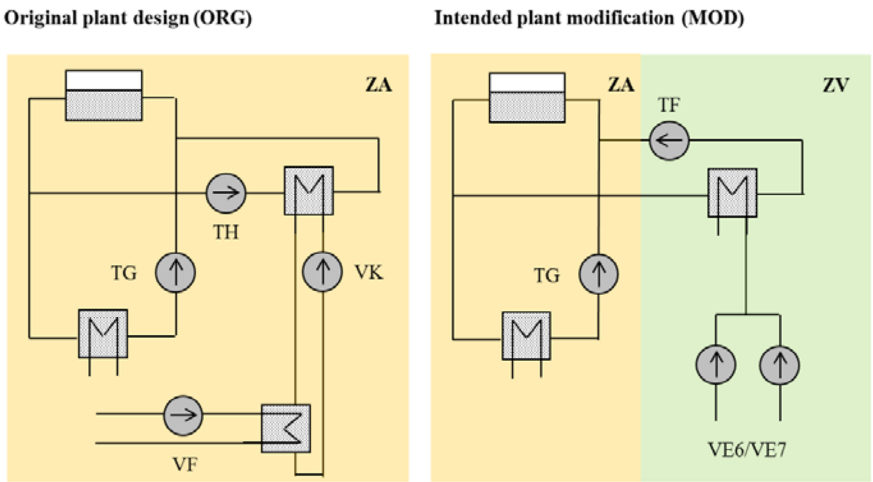
The political decision of decommissioning the NPP Philippsburg, Unit 1 (KKP 1) has necessitated a probabilistic analysis of the long-duration post-commercial plant operational phase. In this phase, the fuel has been removed from the reactor pressure vessel and all fuel assemblies that require cooling are stored inside the spent fuel pool.

The unit was taken off the grid in March 2011; the integral decay heat output in the spent fuel pool is currently less than 1 MW. The plant state of post-commercial shutdown operation is largely characterised by the fact that

- the spent fuel assemblies have been removed from the reactor pressure vessel and stored in the spent fuel pool,
- the majority of the systems needed for power operation are out of service, empty, pressureless and cold as a result of their isolation, and that
- only those systems remain operational that are necessary for this plant state.

The shutdown phase after commercial operation includes measures for the disposal of fuel and media as required by the operating licence and as far as feasible also measures to adapt the plant with a view to reducing post-operational costs and preparing it for the decommissioning and dismantling phase. Post-commercial plant operation ends with granting the first decommissioning and dismantling licence or with granting the licence to establish the safe enclosure.

In the assessment of the post-commercial shutdown operation phase, different plant operational states (configurations of the engineered systems during maintenance work) are being considered. For this purpose, a reference year with specified partial unavailability of such systems is defined.



abbr.	system
ZA	reactor building
ZV	Emergency building
TG	Spent fuel pool cooling system
TH	Residual heat removal system
VK	component cooling system
VF (ESWS)	Essential service water system
TF (IESW)	Independent emergency cooling system
VE6/VE7	ESWS for IECS/ well

Emergency power supply

4 emergency diesel generators

2 independent emergency diesel generators

2 emergency diesel generators

2 independent emergency diesel generators

Figure 2. **Intended plant modification during post-commercial shutdown operation**

For the KKP 1 reference plant, two post-commercial operation configurations of spent fuel pool cooling, ORG and MOD, have been analysed and quantitatively compared (figure 2). They mainly differ in the number of emergency power diesels available and the cooling systems used for the spent fuel pool.

With both configurations, the decay heat of the fuel assemblies is normally removed via the spent fuel pool cooling system TG. The planned modification of the original configuration, abbreviated as MOD, provides for the use of the two cooling trains TF1 and TF2 of the independent emergency system, if the TG system fails or is unavailable according to maintenance. Here, four emergency diesel generators (2 unit diesel generators and 2 diesel generators of the independent emergency system) are provided for emergency power supply. With the ORG configuration, which largely describes the actual plant condition, the two cooling trains TH30 and TH40 of the TH residual heat removal system are used in the event of a failure or unavailability of the TG system. In the case of ORG, six emergency power generators (4 unit diesel generators and 2 diesel generators of the independent emergency system) are provided for emergency power supply. Each of the mentioned cooling trains is able to remove the decay heat from the fuel assemblies all by itself.

The reason for the changed plant configuration are, on the one hand, the more economical future mode of operation and, on the other hand, potential savings by their implementation compared with the expenditure needed for the backfitting measures that would have been necessary with the current configuration. In addition, the safety of the plant with respect to external hazards is enhanced as the current independent emergency systems can be used for spent fuel cooling after the modification.

4. Conducting a Comparative Analysis for the Internal Hazard Fire

The licensee has carried out a probabilistic risk analysis to compare alternatives of spent fuel pool cooling for the longer duration of the post-commercial shutdown phase. The comparison of the risk is based on the annual frequency of fuel damage states (FDF) for both alternatives. The post-commercial shutdown plant state is divided into plant operational states representing typical configurations of the safety system according to maintenance and repair work during a reference year. The probabilistic plant model contains the entire plant operational states and has to be extended such that the fire induced risk can be quantified.

Furthermore, a comprehensive Level 1 full power operation Fire PSA does exist [1], which also means that a suitable partitioning of the plant into individual compartments is available and that the inventory of each of these compartments, including cables, is well known. As the analysis to be carried out is a comparative one, it is possible to adopt the data from the Fire PSA for full power operation wherever it can be assumed that the corresponding values are the same in both alternatives. This is valid, e.g. for the compartment related fire occurrence frequencies and the damage probabilities of those components that have been determined as relevant by means of the fire event trees. Otherwise, the differences between the two alternatives of SFP cooling are reflected in the fault tree models of the systems for residual heat removal.

In order to be able for quantitatively assessing the effects of fire induced component and cable failures on the fuel assembly damage states for the two variants ORG and MOD, fire induced failures are additionally considered for all components based on the event and fault tree models of the licensee. To this end, for each of these components those compartments are screened out, in which a fire may lead to the failure of the corresponding component. This information was taken from the Fire PSA [1].

As an example, the failure of a motor operated gate valve VE72S101 has been analysed. This valve will not open if a technical fault occurs or if a common cause prevents it from opening together with VE62S101. For a fire analysis, an allocation of components to the different compartments is needed (see Chapter 2 for the methodology). The motor operated gate valve VE72S101 is located in room ZV03-03, with the associated cables, power cables as well as I&C

cables, routed through rooms ZV02-03, ZV02-06, ZV02-07K and ZV04-04. Hence the gate of the fault tree describing the failure of the motor operated gate valve is then replaced in the plant model as shown in Figure 3.

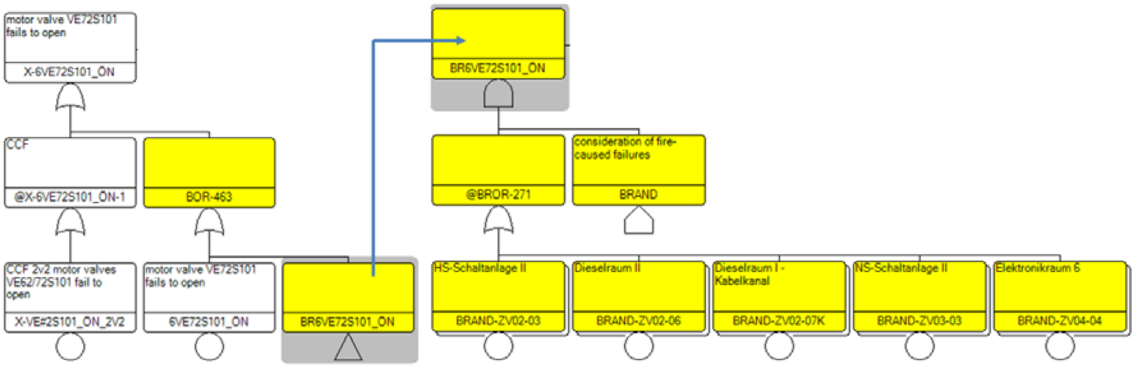


Figure 3. **Extension of the component failure descriptions by fire-induced failures**

The fault trees in the plant model have been extended automatically.

As a rule, fire induced failures are additionally considered for all component failures modelled in the comparative probabilistic risk analysis. The extension of the fault trees has been carried out applying a program developed at GRS [5]. The fire induced failure probabilities are conditional probabilities for the compartment failures, taken from [1].

For the reliability models of fire induced failures to be used, it is distinguished between plant operational states with and without system isolations:

- For plant operational states with system isolations, the fire induced failures are considered for the time period of the respective state.
- For plant operational states without system isolations ("trouble-free post-operational phase"), the fire induced failures are modelled as self-reporting failures with a specific failure rate and repair time. This means that the components in the fire compartment fail with the postulated fire occurrence frequency and on average remain unavailable for the assumed repair period.

In the first case, the assumption is that an intended system isolation will only be carried out if there is no fire at the beginning of the isolation. The second case is based on the assumption that following the occurrence of an incipient fire or following fire damage, components can be repaired or that it is possible to establish a system state that corresponds to the original system state as regards the reliability of SFP cooling.

In comparing the two SFP cooling alternatives ORG and MOD, the following initiating events have been considered: loss of offsite power (LOOP), failure of residual heat removal (RHR) from the spent fuel pool, loss of water from the spent fuel pool, and flooding induced unavailability of the required system functions of the independent emergency systems (IES) in the corresponding IES building. Fires may cause the initiating events LOOP and RHR failure from the spent fuel pool.

The result of the quantitative analyses is that the risk of fuel damage is much lower in case of SFP cooling using MOD.

It has to be mentioned in this context that the estimated risk values do not represent absolute values but should only be only applied for a comparison of the two alternatives. For carrying out a comprehensive and complete Fire PSA to assess the risk for the second alternative, the following steps should be performed in addition:

- Development of fault trees for the initiating events LOOP and spent fuel pool RHR failure (in these fault trees, the initiating event is traced back to fire induced component failures),
- Fire specific analyses for all plant operation states of the post-commercial plant operational phase and the corresponding differences to full power operation (compartment specific fire occurrence frequencies, possibilities of fire propagation, fire extinguishing possibilities, waiting periods´ extension).

Based on a full power operation Level 1 Fire PSA and a comparative probabilistic analysis for plant internal initiating events, it was possible to carry out effectively and in a short time period a comparative fire risk analysis for two alternatives of spent fuel pool cooling.

5. Conclusions

The comparative probabilistic fire risk analysis for the post-commercial shutdown plant operational phase has shown that even if the initiating event fire is taken into account, the spent fuel pool cooling alternative MOD has a considerably lower frequency of fuel damage than the ORG alternative. It thus has been demonstrated that regarding fuel damage states, the system alternative MOD represents a clear improvement over the alternative ORG. This is largely due to the following:

- The cooling trains TF of the independent emergency system in the spent fuel pool alternative MOD have redundant auxiliary service water systems while in the alternative ORG, each TH train has only one train of the VF auxiliary service water system at its disposal.
- Spent fuel pool cooling with the trains of the independent emergency systems is more simply structured compared to spent fuel pool cooling with the TH trains; there is no component cooling system required.
- The connection of the TH trains requires the actuation of a larger number of valves than the alternative using TF trains.

The fact that there is a higher degree of redundancy regarding the emergency power generators in the ORG configuration than in the MOD one does not play any significant role for the result in this case as the contribution of the emergency power generator failures to the overall failure rate is comparatively low.

In line with the purpose of the study, the risk values determined are comparative values and must not be seen as absolute values.

References

[1] Babst S., et al. (2005), Brand-PSA für das Kernkraftwerk Philippsburg, Block 1 im Leistungsbetrieb, GRS-A-3278, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, June 2005

[2] Babst, S., Röwekamp, M., Türschmann, M. (2013), Application of Fire PSA in Case of Modifications for Post-operational Shutdown States, EUROSAFE 2013, Köln, November 2013

[3] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke. Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-358/05, Salzgitter, Oktober 2005

[4] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke. Ergänzungsband zu Methoden und Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Draft, Salzgitter, 2014 (to be published 2014)

[5] Herb, J. (2012), Fault Tree Auto-Generator: How to Cope with Highly Redundant Systems, in: 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference 2012, (PSAM11 ESREL 2012). ISBN: 978-1-62276-436-5, Curran Associates, Inc., Red Hook, NY, 2012