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Fire protection in the operational safety of nuclear installations
Current trends and issues

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DG JRC – Institute for Energy

December 2006

SENUF

Safety of Eastern European Type Nuclear Facilities

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Executive Sumary

This report represents an initial research effort in the field of fire safety according to the EC-JRC action Safelife objectives and in view of the new EC-JRC action named SONIS (Safety of Operating Nuclear Installations) that will be implemented in FP7. The new action will address fire safety issues for systems, structures, components, and particularly for the I&C (instrumentation and Control) and electrical equipment, gathering the available experience, setting up information data bases and developing suitable qualification procedures.
In particular, this report summarizes the state of the art engineering approaches and identifies the challenges posed by large fire scenarios on plant safety, where conventional assessment techniques are not applicable, in view of the development of safety strategies for a safe plant shutdown.
The final set of recommendations and research needs addresses material properties investigation, deterministic and probabilistic safety assessment of systems, structures and components and their capacity to withstand large fires ignited at the site by both internal and external sources.
The content of this report is mainly based on the research results presented at the two following key events in the fire engineering community, where some of the authors also contributed with scientific papers:

- Post SMiRT Conference on Fire safety, Vienna in August 2005
- Fire PRA, Washington in May 2006

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1 Introduction

1.1 Background

Nuclear power plant operation experience over the past three decades indicates that fires at NPPs may constitute a real and significant threat to nuclear safety in addition to the conventional fire hazards to life and property. It is widely acknowledged that in many cases, the risk posed by fires can be comparable to or even exceed the risk from internal events.

Hence, large efforts have been spent internationally to fully understand and analyse the phenomenon of fire and its consequences at NPPs on one hand and to improve NPP design and regulatory requirements to fire safety as well as fire protection technology on the other hand.

The EC well recognises the high relevance of fire safety, among the other programs in the operational safety area: the Direct Action assigned to the JRC under the Framework Programme n°7 (FP7) EURATOM part for the period 2007-2011 explicitly identifies the “Safe Operation” of operating and future plants as the top priority activity in the Nuclear Safety area. Other Policy Documents, such as the Convention of Nuclear safety, the Council resolutions C 128-75 and C172/2-92 on harmonisation of safety practices, and the Green Paper on Energy COM(2000) 769 (final), particularly for what concerns the security of the energy supply, make an explicit reference to the safe plant operation, including the fire safety among the other areas of major concern.
In these documents, after analysis of the experience feedback and of the priorities of the Member Countries, the characteristics of the required action have been identified in the following: operation of a forum of exchange of experience, harmonization of practices among European Countries, application of advanced techniques to safe plant management, etc.

From the technical standpoint, it is recognised that all nuclear facilities are designed in relation to accidental fires; even so, the experience feedback shows that fire scenarios are still among the main contributors to the overall vulnerability of the nuclear installations and it also confirms both high frequency and high risk significance of events in this field.
In the safety assessment of nuclear installations in relation to fire scenarios, the modelling of the scenarios improved significantly over the last years. However, it still represents the largest contribution to the overall uncertainty in the results together with the lack of data on the vulnerability of the SSC (systems, structures and components). In fact failure modes and thresholds for many components are not completely understood in terms of effects from temperature, smoke, aerosols, lack of oxygen, etc.
Moreover, a recent interest in more complex scenarios, such as those related to large fires (i.e. ignited on large areas at the same time), makes the assessment even more difficult, as explosions, mechanical impacts and fire spreading on large areas are combined with the pure thermal effects, due to the size of affected areas and the

probable failure of the separating barriers. Particularly in these cases, the simplistic assumptions of failure of all the equipment in the area affected by the large fire becomes too conservative and unacceptable, demanding for more refined qualification procedures.

Therefore the EC-JRC action named SONIS (Safety of Operating Nuclear Installations) that will be implemented in FP7 aims at filling this gap for systems, structures, components, and particularly for the I&C (instrumentation and Control) and electrical equipment, gathering the available experience, setting up information data bases and developing suitable qualification procedures.

This report represents an initial research effort in the field of fire safety according to the EC-JRC action Safelife.
In particular, the report summarizes the state of the art engineering approaches and identifies the challenges posed by large fire scenarios on plant safety, where conventional assessment techniques are not applicable, in view of the development of safety strategies for a safe plant shutdown.
The final set of recommendations and research needs addresses material properties investigation, deterministic and probabilistic safety assessment of systems, structures and components (SSC) and their capacity to withstand large fires ignited at the site by both internal and external sources.
The content of this report is mainly based on the research results presented at the two following key events in the fire engineering community, where some of the authors also contributed with scientific papers:

- Post SMiRT Conference on Fire safety, Vienna in August 2005
- Fire PRA, Washington in May 2006

1.2 Objective

The objective of this report is to provide insights into some current trends and issues in nuclear power plant fire risk assessment, in order to support further actions under the EC Framework Program n°.7 (FP7), possibly under the new EU/JRC action SONIS.
Based on the judgment and experience of the author, the paper does not intend to present a complete study of the state of the art of NPP fire protection and risk assessment and related research. Rather, the intention is to give a brief overview on methods and tools that are currently in general use in the NPP fire safety community. Particular attention is paid on some issues where the engineering practice is not considered to be on a level comparable to other areas of nuclear safety assessment.

1.3 Document structure

Section 2 describes the main design assumptions and the reference fire scenarios. Section 3 covers the fire risk concept, presents key concepts of NPP fire protection and describes the state of the art of current fire risk analysis methods. Section 4 discusses the evaluation of fire fragilities for systems, structures and components. Section 5 provides a brief overview on TACIS projects funded in this field, as additional background to the evaluation of priorities and research needs in this area.

Section 6 is dedicated to the identification of areas where the current state of the art of fire risk assessment is not yet at a level comparable to other NPP safety assessment domains and further research is hence necessary. Finally Section 7 provides some conclusions on the implementation and further research needs in the fire risk assessment of nuclear power plants, providing also a tentative research work programme.

2 Reference fire scenarios

Fires are an important risk at NPPs as a single fire may incapacitate multiple diverse and redundant engineered systems triggering a common cause failure that challenges nuclear safety. This is in particular the case at NPPs designed to earlier standards where the original fire protection concept was based on normal industrial practices with the emphasis on protecting from property losses and no or little attention to the specific requirements of nuclear safety.

Fire-related incidents or events occur relatively frequently at NPPs. Alone in the history of US nuclear power industry, more than 2500 events have occurred and been recorded in the EPRI fire events database and it is estimated that the frequency of significant fire events is in the order of once in ten reactor-years. Of course, most of the recorded events have been irrelevant for nuclear safety. However, a number of fires have created serious nuclear safety challenges and in some cases, serious nuclear accidents have been avoided by only a very thin margin, sometimes even rather by luck than planned technical response. Core damage precursor fires have occurred, among others at the following NPPs: Browns Ferry (US, 1975), Greifswald (GDR, 1975), Beloyarsk (USSR, 1978), Armenia (USSR, 1982), Vandellos (Spain, 1989), Chernobyl (Ukraine, 1991) and Narora (India, 1993)

Not all large nuclear power plant fires are significant from a public safety point of view, nor are all safety significant fires large. Differences in such details as the routing of key electrical cables, the separation and orientation of important cable trays, the fire protection scheme used for a particular compartment, and the procedures employed by plant operators in response to a fire can dramatically alter the risk significance of real and hypothesised fires.

In general, all nuclear facilities are designed in relation to accidental fire scenarios, under certain assumptions on their initiation and development mechanisms. However, conventional fire hazard analysis is based on the hypothesis of the presence of combustible materials in the buildings and limited number of simultaneous sources of fire. In addition, conventional fire safety assessment relies upon the presence of mitigation measures and fire related operational procedures.
The operating experience in the nuclear facilities all over the world (see for example [27]) shows that these assumptions may not be always realistic, particularly in relation to the fire initiation. In many plants fire is still among the most common event initiator and its risk contribution to any Core Damage Frequency of nuclear plants is still one of the most significant.
In addition to that, recent emphasis on large fire scenarios implies the need of the safety assessment of the facilities in relation to large induced fire, also originated from external sources and/or ignited inside the structures by fire loads transferred from the

outside. These scenarios show special characteristics, not addressed by the current engineering practice for the design of nuclear installations, such as: involvement of large areas, spreading from outside the buildings, breach in fire barriers, reduced availability of suppression measures and emergency actions, combination with explosions, smoke and mechanical impacts.

3 Summary review of the safety assessment practice in relation to fire safety

3.1 The Defence-in-Depth approach applied to fire protection

NPP fire protection concept is (or should be) based on three main layers of defence in depth, complemented by appropriate operational procedures to manage possible fire-induced-losses:

- Prevent fires from being ignited
- Detect and extinguish any occurring fires rapidly to limit damage
- Prevent spreading of those fires that have not been extinguished to minimise potential fire effects on key plant systems and functions

Fire prevention is achieved by minimising fire loads through appropriate material choices and good housekeeping to minimise transient fire loads, and minimising ignition sources through regular predictive maintenance and in-service inspections of equipment as well as appropriate hot work procedures and permits.

Active fire protection measures such as detection and suppression systems serve the purpose of rapidly limiting the damage caused by any occurring fires. The design should be balanced to achieve rapid detection and suppression e.g. by automatic systems whilst avoiding false or spurious alarms and release of suppression agents which may themselves cause some damage to equipment.

Passive fire protection measures serve the purpose of preventing fire spreading and minimising the impact on key systems and functions. Each train of redundant safety systems should be physically separated from each other as well as normal operating systems. This is achieved by establishing fire compartments i.e. plant areas completely surrounded by fire barriers of a defined fire resistance rating. These comprise e.g. fire rated walls, ceilings, doors, penetrations and dampers as well as local fire insulation. Fire cells within fire areas are separated spatially or by function. If safety systems are separated from each other only by distance, the fire protection should be complemented by active fire protection means.

Fires should be taken into account in normal and emergency operating procedures so as to manage possible fire-induced losses. NPP operators should be trained for fire events in order to establish proper co-ordination with the plant operation and fire-fighting efforts. Plant fire brigade training and fire extinguishing plans should take into account risk-informed aspects.

3.2 Deterministic Fire Hazard Analysis

It is standard practice in many countries operating nuclear power plants to rely on a deterministic Fire Hazard Analysis (FHA).

A FHA is performed to evaluate the consequences of potential fire hazards and to develop appropriate fire protection systems and features.

According to [20], the FHA should cover all relevant areas of the plant to clearly demonstrate there is a sufficient level of protection in accordance with the defence in depth concept described in Section 3.1. It should be performed by qualified fire protection and reactor systems engineers and include the following:

- The evaluation of physical construction and layout of buildings and equipment (including electrical cables) within fire compartments or fire cells.
- An inventory of combustibles, including maximum transient combustibles within each fire compartment or cell.
- A description of fire protection equipment, including detection systems and manual and automatic extinguishing systems in each fire compartment or cell.
- An analysis to assure a single fire event (in any compartment or cell) cannot impair required safe shutdown functions or result in the uncontrolled release of radioactive contamination to the environment. The analysis should use recognized technology as a basis for conclusions with consideration given to:
    - fire growth and heat released from a design basis fire and
    - the potential adverse effect operating fire suppression systems can have on safe shutdown functions.
- An analysis of irradiated fuel storage areas.

When applied to the assessment of the existing facilities, the FHA can identify specific areas where levels of fire protection are inadequate and where corrective measures are necessary and provide a technical justification for any deviations from the recommended practices for which no corrective measures are taken.

3.3 Probabilistic Fire Safety Assessment

3.3.1 State of the art in the Fire PRA/PSA methodology

Fire-PRA (Probabilistic Risk Assessment) [4] has been performed in numerous countries to different levels of depth and coverage and in many countries it can be used as an element to support risk-informed decision making in addition to the traditionally required deterministic analyses.

The current state of the art for fire PSA (Probabilistic Safety Assessment) is well described in the CSNI technical opinion paper 1 [5]: "The general approach for performing a fire PSA is little changed from that used in the earliest commercial nuclear power plant fire PSA studies (performed in the early 1980s). As part of this approach, potentially important scenarios are identified through a consideration of the fire hazards (ignition sources and fuels) in each plant fire area and of the plant equipment (including electrical cables) that may be damaged by a fire. Of particular interest are fire initiated scenarios involving the triggering of a plant transient (an "initiating event") and the degraded response of plant systems and operators. The scenario frequencies are quantified by estimating the frequency of fire initiation, the conditional probability of fire-induced damage to critical equipment given the fire, and the conditional probability of core damage given the specified equipment damage. "

This method of quantifying the fire-induced core damage frequency is commonly expressed by the following equation:

$$CDF = \sum_i \lambda_i \sum_j p_{ed,j|i} \sum_k p_{CD,k|i,j} \qquad (1)$$

,where

λ_i is the frequency of the fire scenario i

$p_{ed,j|i}$ is the conditional probability of damage to critical equipment set j given the occurrence of fire scenario i

$p_{CD,k|i,j}$ is the conditional probability of core damage due to plant response scenario k given fire scenario i and damage to critical equipment set j

Note that the first term quantifies the first layer of defence in depth ("prevent fires from being ignited"). The second term addresses the issues of fire growth, detection, suppression, and component damageability and thus quantifies the second and third layers of defence in depth ("detect and extinguish any occurring fires rapidly to limit damage" and "prevent spreading of those fires that have not been extinguished to minimise potential fire effects on key plant systems and functions"). The third term addresses the unavailability of equipment unaffected by the fire and/or operator failures.

The method is further explained in the CSNI technical opinion paper 1 [5] as follows: "(Post-core damage issues, e.g., containment failure and radioactive material release, can also be treated, but this has only been done in a limited number of studies.) Typically, the fire initiation frequency is estimated using a simple statistical model for fire occurrences; the likelihood of fire damage is estimated using combinations of deterministic and probabilistic models for the physical processes of fire growth, detection, suppression, and equipment damage; and the likelihood of core damage is estimated using conventional PSA systems and accident progression models. Because of the large number of scenarios that need to be considered, the assessment process is iterative and involves the extensive use of screening techniques to focus analysis resources on the most important scenarios."

Widest use of Fire PRAs is made in the US, where the U.S. Nuclear Regulatory Commission (NRC) has amended its fire protection rule in Title 10 of the Code of Federal Regulations Section 50.48 (10 CFR 50.48) [13] to allow nuclear power plant licensees to voluntarily adopt a risk-informed and performance-based rule. A reactor licensee is permitted to use the fire protection requirements contained in the National Fire protection Association (NFPA) Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 edition [2], with exceptions, as an alternative to complying with the deterministic requirements of 10 CFR 50.48(b) or the licensee's fire protection license condition.

The American Nuclear Society has issued a draft standard presenting requirements for a fire PRA that can be used for risk-informed decision making [1]. All relevant areas of a fire-PRA, complementing a level 1 (and level 2 as applicable) power operation internal events PRA, are covered. Three Capability Categories, presenting criteria to the scope and level of detail, plant-specificity and realism of the analysis, are defined. The logic and structure of the ANS draft standard follow those of the EPRI/NRC-RES Fire PRA Methodology [3], where state of the art methods, tools and data for conducting a fire PRA are given. This methodology is discussed in the following section in more detail.

3.3.2 Summary of the NRC/RES/EPRI Fire PRA methodology

The NRC/RES/EPRI Fire PRA methodology [3] is well developed and comprehensive, presenting state of the art methods, tools and data for conducting a fire PRA that can be used for risk-informed decision making. All relevant areas of a fire-PRA, complementing a level 1 (and level 2 as applicable) power operation internal events PRA, are covered in the following 16 tasks (extract from NUREG/CR-6850) [3]:

Plant Boundary Definition and Partitioning (Task 1). The first step in a Fire PRA is to define the physical boundary of the analysis, and to divide the area within that boundary into analysis compartments.

Fire PRA Component Selection (Task 2). The selection of components that are to be credited for plant shutdown following a fire is a critical step in any Fire PRA. Components selected would generally include all the components credited in the 10 CFR 50 Appendix R post-fire safe shutdown (SSD) analysis. Additional components will likely be selected, potentially including all components credited in the plant’s internal events PRA. Also, the proposed methodology would likely introduce components beyond either the 10 CFR 50 Appendix R list or the internal events PRA model. Such components are often of interest due to considerations of combined spurious actuations that may threaten the credited functions and components.

Fire PRA Cable Selection (Task 3). This task provides instructions and technical considerations associated with identifying cables supporting those components selected in Task 2. In previous Fire PRA methods (such as EPRI FIVE and Fire PRA Implementation Guide) this task was relegated to the SSD analysis and its associated databases. This document offers a more structured set of rules for selection of cables.

Qualitative Screening (Task 4). This task identifies fire analysis compartments that can be shown to have little or no risk significance without quantitative analysis. Fire compartments may be screened out if they contain no components or cables identified in Tasks 2 and 3, and if they cannot lead to a plant trip due to either plant procedures, an automatic trip signal, or technical specification requirements.

Plant Fire-Induced Risk Model (Task 5). This task discusses steps for the development of a logic model that reflects plant response following a fire. Specific instructions have been provided for treatment of fire-specific procedures or pre-plans. These procedures may impact availability of functions and components, or include fire-specific operator actions (e.g., self-induced-station-blackout).

Fire Ignition Frequency (Task 6). This task describes the approach to develop frequency estimates for fire compartments and scenarios. Significant changes from the EPRI FIVE method have been made in this task. The changes generally relate to use of challenging events, considerations associated with data quality, and increased use of a fully component based ignition frequency model (as opposed to the location/component-based model used, for example, in FIVE).

Quantitative Screening (Task 7). A Fire PRA allows the screening of fire compartments and scenarios based on their contribution to fire risk. This approach considers the cumulative risk associated with the screened compartments (i.e., the ones not retained for detailed analysis) to ensure that a true estimate of fire risk profile (as opposed to vulnerability) is obtained.

Scoping Fire Modeling (Task 8). This step provides simple rules to define and screen fire ignition sources (and therefore fire scenarios) in an unscreened fire compartment.

Detailed Circuit Failure Analysis (Task 9). This task provides an approach and technical considerations for identifying how the failure of specific cables will impact the components included in the Fire PRA SSD plant response model.

Circuit Failure Mode Likelihood Analysis (Task 10). This task considers the relative likelihood of various circuit failure modes. This added level of resolution may be a desired option for those fire scenarios that are significant contributors to the risk. The methodology provided in this document benefits from the knowledge gained from the tests performed in response to the circuit failure issue.

Detailed Fire Modeling (Task 11). This task describes the method to examine the consequences of a fire. This includes consideration of scenarios involving single compartments, multiple fire compartments, and the main control room. Factors considered include initial fire characteristics, fire growth in a fire compartment or across fire compartments, detection and suppression, electrical raceway fire barrier systems, and damage from heat and smoke. Special consideration is given to turbine generator (T/G) fires, hydrogen fires, high-energy arcing faults, cable fires, and main control board (MCB) fires. There are considerable improvements in the method for this task over the EPRI FIVE and Fire PRA Implementation Guide in nearly all technical areas.

Post-Fire Human Reliability Analysis (Task 12). This task considers operator actions for manipulation of plant components. The analysis task procedure provides structured instructions for identification and inclusion of these actions in the Fire PRA. The procedure also provides instructions for estimating screening human error probabilities (HEPs) before detailed fire modeling results (e.g., fire growth and damage behaviours) have been developed. Estimating HEP values with high confidence is critical to the effectiveness of screening in a Fire PRA. This report does not develop a detailed fire HRA methodology. There are a number of HRA methods that can be adopted for fire with appropriate additional instructions that superimpose fire effects on any of the existing HRA methods, such as SHARP, ATHEANA, etc. This would improve consistency across analyses i.e., fire and internal events PRA.

Seismic Fire Interactions (Task 13). This task is a qualitative approach to help identify the risk from any potential interactions between an earthquake and fire.

Fire Risk Quantification (Task 14). The task description provides recommendations for quantification and presentation of fire risk results.

Uncertainty and Sensitivity Analyses (Task 15). This task describes the approach to follow for identifying and treating uncertainties throughout the Fire PRA process. The treatment may vary from quantitative estimation and propagation of uncertainties where possible (e.g., in fire frequency and non-suppression probability) to identification of sources without quantitative estimation, where knowledge of a quantitative treatment of uncertainties is beyond the state-of-the-art. The treatment may also include one-at-a-time variation of individual parameter values to determine the effect on the overall fire risk (sensitivity analysis).

Fire PRA Documentation (Task 16). This provides suggestions for documenting a Fire PRA.

3.4 Comparison between probabilistic and deterministic approaches

A useful comparison between the two discussed techniques is provided in the following table 1 from [5]. Useful references are also available in [18-26].

Table 1 – Comparison between deterministic and probabilistic approaches in fire safety assessment

Comparison of Deterministic and PSA Fire Analysis		
Issue	***Deterministic Approach***	***Fire PSA Approach***
Extent of equipment/ cable damage	Generally assumes all equipment in fire compartment will be damaged or fire spread will be limited by physical separation and /or installed suppression systems	Uses fire modeling to determine extent of damage from specific sources
Likelihood of fire	Assumes fire may occur regardless of sources present	Evaluates component/activity-based fire frequencies for specific plant locations based on generic and/or plant-specific experience
Coincident equipment failures	Assumes equipment unaffected by the fire will be available for plant shutdown	Considers random failures of unaffected equipment coincident with fire damage
Operator reliability	Assumes operators will take actions directed by procedures having demonstrated minimum time, instrumentation and access is available	Considers likelihood of potential operator error given stress factors imposed by particular fire, spurious alarms or lack of alarms, existing operating

		procedures not adapted to the situation induced by fire and activity-related conditions (e.g. degraded instrumentation, smoke, time limitations and training)
Offsite power and other non-safety systems	Often assumes non-safety related components and services (e.g. offsite power and main feedwater) are unavailable	Only assumes components and services are unavailable if shown to be damaged by the fire; otherwise considers random failure probability coincident with the fire
Fire protection systems	Has specific requirements regarding installation and operability depending upon fire hazard; if these are met analyses assume system is effective	Accounts for random failure of automatic and manual fire detection and suppression systems depending upon specific types of system installed and fire fighter response during drills; also addresses random barrier element failure probabilities

4 Equipment qualification in relation to fire hazard

In recent years a number of scientific papers addressed the component reliability in relation to fire scenarios, as answer to a growing demand from the industry for the improvement of the reliability of the overall safety assessment processes. In fact it is generic convincement that one of the weakest part of the plant safety assessment in relation to fire is the evaluation of the component capacity to withstand the different effects induced by these scenarios.
In particular, the component qualification issue becomes very relevant: the failure modes for many safety related components are not completely understood in terms of effects from temperature, smoke, aerosols, shock waves, lack of oxygen, etc. While some data are available for the fire capacity of walls, doors, cables, dampers, etc., I&C and electrical systems are largely uncovered by qualification databases.

Some research contributions are summarised in the following:

1) Many research programs addressed the effects of smoke on control electronics used in programmable automation circuitry in NPPs. They were studied experimentally and theoretically. Relevant physical parameters of smoke, and electrical performance of circuitry were measured online. It was noticed that the insulation resistance decreased three orders of magnitude due to soot deposits on surfaces on uncoated circuitry. Quantitative models for smoke exposure and deposition on surfaces were proposed based on existing physical models of aerosols (see for example [11])
2) Other studies addressed the reliability of safety systems (especially emergency shutdown systems) in case of fire. The studies tried to evaluate how long certain critical instruments and machinery would operate after their long term ambient temperature limits have been exceeded. For exploring the problem experimental measurements were carried out. Damages and failures of the equipment were observed. Simple theoretical models were derived (see for example [10])
3) A specific group addressed the failure probability of redundant cables in cable tunnel fires, and the failure and smoke filling probabilities in

electronics room during an electronics cabinet fire. Modelling parameters and actual facility properties that have the most influence on the results were discussed and models provided (see for example [8,9])

In conclusion, it is believed that the gathering of available fire qualification data and the development of new qualification procedures, when such data are not available, could tremendously improve the quality of the overall safety assessment of the nuclear installations to both accidental and malevolent fire dominated scenarios.

5 Experience feedback - Fire Safety in CIS countries

During more than a decade's efforts, the TACIS programme financed by the European Commission have made a significant contribution to analyse and improve the safety of nuclear power plants in the former Soviet countries.

The TACIS programme for improving nuclear safety was developed in accordance with the G7 strategy adopted in Munich in 1992, and reflected the International Atomic Energy Agency (IAEA) classification of design and operating risks regarding Soviet-designed nuclear reactors. Fire was identified, also in the light of operational experience and exchange between Eastern and Western experts, as one of the prime safety concerns at Soviet-designed NPPs.

Fire safety was given high priority in the program and was covered by many projects. Projects worth more than 11 M€ were implemented in the areas of on-site assistance, design safety and regulatory assistance, consisting of improvements of passive and active fire protection measures as well as various analyses and studies. In terms of total budget, roughly 30 % was dedicated to improving active fire protection measures, some 55 % to passive protection measures and 15% to studies [12].

The project programming and identification performed in the frame of the Master Planning in the nineties took fully into account internationally available information on the fire safety deficiencies of the different Soviet-designed NPPs and recommendations for improvement such as those presented in the IAEA reviews [14-17]:

- RBMK: "... Similar to other nuclear power plants, fire protection and detection, and fire suppression capability are important safety issues. Some improvements have already been implemented, but safety concerns remain. For example, the practice of using plastic floor covering in the plant adds to the fire hazard because it can generate dense and toxic smoke in case of fire..."

- VVER-440/230: "... A full scope hazards analysis starting with fire and flooding (as the most common ones) is needed to identify and correct the numerous and significant sources of common cause failures detected during the safety reviews..."

- VVER-440/213: "... A systematic fire hazards analysis for each area of every VVER-440/213 plant is needed. This analysis should identify the weak points of fire barriers, show the need to separate redundant trains of safety importance and justify the acceptability of redundant train separation by distance...”

- VVER-1000: “... A comprehensive fire protection programme has not been established in most VVER-1000. The fire safety standards used for the design are rather old. There are deficiencies in the fire detection and suppression systems, the fire barriers and the cable spreading room design...”

The IAEA review ranked fire protection of VVER-440/230 to Category III[1] on the issues of fire protection analysis, fire protection equipment and inspection of fire protection equipment. For VVER-1000 plants, passive fire protection and protection of cable spreading rooms were ranked Category III and systematic fire hazards analysis Category II.
Consistently, most of the TACIS project supplied state-of-the-art detection and suppression systems, supplied better isolation barriers (doors, dampers, cable coating, etc.), improved the fire resistance of metal structures (cable rooms, turbine roof, etc.), reduced some fire loads (change of floor coating), supported the development of fire risk assessment for the nuclear island and the diesel generator buildings, provided training to personnel, reviewed fire PSA studies.

Besides the wider concern related to the VVER and RBMK plants, a number of projects were also dedicated to specific issues of sodium cooled BN type reactors. Support was also given to the Russian Regulatory Authority in the review of the fire-PSA of a VVER-1000 plant (Balakovo NPP Unit 4).

Project description sheets for TACIS projects can be found on the EC-JRC-IE/TSSTP website in the section dedicated to the dissemination of project results [7].

Some of the projects in the fire safety area are particularly relevant in the context of this report, notably the project R2.08/96 "Fire Risk Assessment for All Types of Reactors". The project R2.08/96 was carried out in 1999-2000. FRA methodology was developed, including six guidance documents for the subtasks of a fire-PSA:
- Building analysis
- Room analysis
- Determination of transient events and systems with reduced functionality due to a fire
- Probabilistic assessment of fire protection equipment
- Definition of fire related sequences for plant specific PSA
- Quantification of event sequences

The FRA methodology was applied at a VVER-440/V230 reference plant (NvNPP 3) and recommendations for the application at all reactor types were issued. The project showed that the standard PRA/PSA methodologies, mainly modified from the original US versions, could be well generalised and applied to VVER plants, but they need to

[1] IAEA definitions for Categories II and III:
”Issues in Category II are of safety concern. Defence in depth is degraded. Action is required to resolve the issue“.
”Issues in Category III are of high safety concern. Defence in depth is degraded. Immediate corrective action is necessary. Interim measures might also be necessary“.

be extended to include an assessment of the effects of fire on safety related systems, including cabling, which are very specific to the VVER design.

The generic feedback from all the above mentioned projects was that fire safety still needs research and engineering: among the other tasks, a better understanding of the component failure modes and their fragilities in relation to fire was a constant recommendation.

6 Generic issues and research needs

Use of fire-PSA methodology and its general principles are today well established both in the industrial and regulatory side. But the quantification of fire-induced core damage frequency still involves significant uncertainties. Use of sophisticated CFD (computational fluid dynamics) fire codes to model the fire development and impact on systems, structures and components, together with Monte Carlo-simulation to handle parameter uncertainty is current state of the art and is becoming accessible to analysts in the NPP fire safety community.

In particular, modelling of fire development and effects is in constant progress and relatively good validation results are obtained for known source terms and pool fires. Models for predicting fire development in solids (particularly cables in the NPP context) remain insufficient.

In the field of the assessment techniques, in order to bring fire risk assessment to an accuracy level comparable to other PSA domains, further research into quantitative fire analysis methodology is necessary with respect to both deterministic and probabilistic models. Methods and tools used in different stages of FRA need further research in the following areas:

- FRA data and its uncertainties
- Modelling of fire ignition and spreading, including the generation and spreading of smoke and its impact on the performance of plant personnel
- Modelling of impact of fire effects (smoke and heat) on electric and electronic equipment
- Modelling of the reliability of fire detection and extinguishing systems
- Assessment of operational fire-fighting
- Reliability of passive systems

However, the most urgent need is usually associated to the availability of basic data used to feed the assessment methods: these data are often of poor quality and not plant specific. Available NPP fire event data are currently incomplete, not suitable for statistical inference and the reporting bases are not consistent. Hence the usability of such data for quantitative assessment, e.g. fire frequencies, performance of detection and extinguishing systems etc. is limited. In lack of detailed fragility values, input for PSA very often assumed on the basis of very simplistic hypotheses.

More specifically, real-scale fire behaviour of fire-resistant (IEC 331) or fire-retardant (IEC 332-3) cables is subject to debate. For example, the designer of Olkiluoto-3 (EPR) considers no fixed extinguishing systems necessary, as FRNC cables (Flame-

Retardant Non-Corrosive) will be consistently used. But the Finnish regulatory authority STUK will require further justification before accepting this.

The NRC/RES/EPRI Fire PRA methodology, whilst being comprehensive and extensively documented, is based on US data and plant designs. Thus, its full applicability to European NPPs is not necessarily guaranteed. In certain areas like circuit analysis, it appears that differences in e.g. I&C systems may have quite significant effects on the analysis effort and outcome.

7 Proposal for future actions

A preliminary proposal for project tasks in the field of fire safety is described in the following, consistently derived by the evidences provided in the chapters above.

Task 1: Identification of the fire related parameters associated to the design basis scenarios for accidental and malevolent events. Identification of the SSCs where fire vulnerabilities are important for the overall safety evaluation of the plants. Detailed analysis of the potential scenarios at the component level induced by fire. The component may show one of the following:

- ignite and burn
- increase the fire hazard
- support the fire spread to adjacent areas
- fail after a certain time of exposure and/or burning

The time of failure may be different from the classified fire resistance time. This is due to the fact that the real fire development is different from the material fire test scenario. Especially the radiation of spill fires and material fire tests may be completely different. Also the threshold values of SSCs may depend on other parameters such as:

- surface temperatures;
- radiation effects; e.g. due to flame impingement
- flows of air and fire gases adjacent to the SSCs
- preheating of SSCs in adjacent areas before the direct fire exposure occurs
- mechanical stresses and deformations due to thermal strain, restraint forces and thermal bending
- impact of fire suppression agents

Moreover, the component failure may be due to the combined effects of impact, blast and fire and/or smoke. Also the presence of fire suppression system (possibly damaged by associated mechanical effects) should be addressed.
Task 1 will end with the definition of the boundary conditions for each component qualification.

Task 2: Gathering and processing of the available data and procedures on equipment qualification either by test or by analysis for nuclear installations in all operational conditions (harsh and mild environment, design basis accident, severe accident, etc.). The scope will include structures (wall, doors, locks), components (mechanical) and electrical and I&C equipment. The qualification will be investigated in relation to temperature, radiation effects, smoke, aerosols and all other environmental variables that the analysis of the design basis scenarios includes. Data will be collected at the major contractor companies, components suppliers certification bodies, both in nuclear and non nuclear oriented organisations.

Task 3: Development of a data base of fire vulnerabilities, designed in agreement with the requirements collected in Task 1

Task 4: Population of the data base with the available data, definition of the dissemination policy to the interested parties, regulation of access and identification of the clearing house.

Task 5: Comparison of available qualification procedure for fire relevant effects. Development of qualification procedures for the components and equipment where no data and procedures are available.

Task 6: Development of a case study on a selected safety related system impacted by a large fire scenario, with evaluation of the benefits gained with the use of a qualification database.

The method of work will rely on a voluntary network of interested nuclear utilities and component suppliers, integrated by original research carried out by EC-JRC-IE staff.

This project part is expected to deliver the final results in some years time, while Task 1, 2 and 3 should be completed in the first year.

The expected outcome could be summarised as in the following:

1. State of the art report on the equipment qualification to fire induced effects
2. Data base with available fire qualification data and links to other available data, ready for use in the safety assessment
3. Procedures for equipment qualification to fire induced effects

The data base and the qualification procedures may be disseminated under special conditions and in agreement with a suitable security policy.

8 Conclusions

The report discussed the main assumptions used in the safety assessment of the nuclear facilities in relation to fire scenarios.
New evidences from the experience feedback suggested reviewing the traditional scenarios used in the design.

Moreover, the analysis of the available engineering methodologies and simulation tools clarified that the engineering community has now available tools for the simulation of the scenarios and of its development (including 3D CFD codes in transient mode), even if still affected by many heavy limitations that need to be addressed in further research programs.
However, it is generic convincement that one of the weakest part of the plant safety assessment is still represented by the evaluation of the component capacity to

withstand the different effects induced by these scenarios. The component qualification issue becomes very relevant: the failure modes for many safety related components are not completely understood in terms of effects from temperature, smoke, aerosols, shock waves, lack of oxygen, etc. While some data are available for the fire capacity of walls, doors, cables, dampers, etc., I&C and electrical systems are definitely uncovered by qualification databases.
This second area will be specifically addressed by the future research program at the JRC-IE, and particularly in the coming project SONIS funded under FP7.

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10 List of abbreviations

CFD	Computational Fluid Dynamics
EPR	European Pressurized Reactor
EPRI	Electric Power Research Institute (USA)
FHA	Fire Hazard Analysis
FRA	Fire Risk Assessment
FIVE	Fire Induced Vulnerability Evaluation
FRNC	Flame Retardant Non-Corrosive [cable]
HEP	Human Error Probability
HRA	Human Reliability Analysis
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
NEA	Nuclear Energy Agency [of OECD]
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NvNPP	Novovoronezh Nuclear Power Plant [Russian Federation]
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment (equivalent to PRA)
SSC	Systems, Structures and Components

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Abstract
This report summarizes the state of the art engineering approaches in the field of fire protection for nuclear installations and identifies the challenges posed by large fire scenarios on plant safety, where conventional assessment techniques are only partially applicable, in view of the development of safety strategies for a safe plant shutdown.
The final set of recommendations and research needs addresses material properties investigation, deterministic and probabilistic safety assessment of systems, structures and components and their capacity to withstand large fires ignited at the site by both internal and external sources.

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