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Upgrading of fire safety in nuclear power plants

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FOREWORD

The lessons learned from experience in nuclear power plant operation indicate that fires in nuclear power plants pose a real threat to nuclear safety and that their significance extends far beyond the scope of a conventional fire hazard.

Considerable progress has been made over the past two decades in the design and regulatory requirements for fire safety, in fire protection technology and in related analytical techniques. Substantial efforts have been undertaken worldwide to implement these advances in the interest of improving fire safety in both new and operating nuclear power plants.

Particular attention is being given to those nuclear power plants that were designed and constructed according to earlier fire protection standards. Systematic examination of the fire safety of these plants is needed in order to identify the existing deficiencies and to implement appropriate corrective measures.

To assist in these efforts, the IAEA has initiated a project devoted to fire safety. The project, which commenced in 1993, concentrated mainly on the development of guidelines for examining, item by item, the adequacy of overall fire safety arrangements for a plant.

This Symposium was intended to provide further assistance in enhancing the fire safety of operating nuclear power plants. It served as a forum for the exchange of practical experience in the systematic assessment of fire safety at nuclear power plants and in the backfitting process.

Various aspects of the upgrading of fire safety at nuclear power plants were discussed during the Symposium, the second of its kind organized by the IAEA. The Symposium covered all relevant elements of the upgrading process: identification of fire safety related deficiencies, the search for the most appropriate corrective measures, and implementation of selected engineering or organizational solutions. Various reviews of fire protection measures, a systematic analysis of fire safety at nuclear power plants, reporting and assessment of fire related incidents and related databases, the regulatory approach to fire safety backfitting, and the lessons learned from the implementation phase were the most important topics covered.

The IAEA is grateful to all those who helped to prepare the meeting, define its scope and structure, select the papers and chair the sessions. It is hoped that the Proceedings will constitute an important source of information to all those concerned with reducing fire risk at nuclear power plants.

EDITORIAL NOTE

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FIRE SAFETY REVIEWS

(Session 1)

Chairperson

U.H.S. SCHNEIDER
Austria

Key Issues Paper

XA9847499

THE NECESSITY OF PERIODIC FIRE SAFETY REVIEW

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1. INTRODUCTION

Nuclear power is clearly a very important commodity in the global community. And, as the 1986 Chernobyl disaster demonstrated so clearly, fire events in nuclear plants can have far-reaching consequences well beyond the plants' physical boundaries. We all share a grave responsibility to ensure that a minimum level of fire safety is provided and maintained in nuclear power plants. This responsibility can best be met by the continual, periodic monitoring of fire safety measures within the plants.

2. DETERMINING FIRE SAFETY LEVELS

Because each plant's resources are limited, the resources available for fire safety efforts must be allocated wisely. Various methods are available to the practicing fire safety engineer for determining how much fire safety is needed. One can approach the problem using basic engineering judgment. This subjective approach can result in wide variation in the level of fire safety achieved, depending on the experience of the engineer and the accuracy of his/her judgments. This is the approach used by a number of plant designers years ago, before fire was identified as an event with significant potential for affecting nuclear safety. The reliance on individual engineering judgment ultimately resulted in levels of fire safety ranging from excessively conservative in some specific plant areas, to a fair overall level in some nuclear plants, to almost no fire safety in other plants.

In consequence, regulatory authorities have developed requirements to establish minimum levels of fire safety which must be provided in nuclear plants. Some of these guidelines are broad, performance-based regulations while others are more prescriptive and offer few choices for the designer. Specific prescriptive-type codes and standards such as those published by the National Fire Protection Association, the Loss Prevention Council, and others offer detailed requirements on the design and arrangement of specific elements of fire safety once a decision has been made to install passive measures or active fire extinguishing systems.

More and more frequently during the past two decades, systematic fire safety analyses have been performed to determine the required level of fire safety in nuclear plants. The analysis often takes the form of a deterministic-type fire hazard analysis (FHA). This comprehensive document can be either qualitative or quantitative in nature; usually it is a combination of the two approaches. Currently, fire safety analyses are being approached using the fire probabilistic safety assessment (PSA) method. This method, which identifies dominant risk contributors, often is used to supplement a deterministic FHA. It can be used as a means for comparing options for risk reduction based on the probability of a specific initiating event leading to a fire of a magnitude sufficient to result in core damage. By screening out less significant events, this method focuses attention on those events which have the highest probability of affecting plant nuclear safety. The broad goal of each method is to ensure nuclear safety for the plant. As the analytical methods in use increase in sophistication and level of detail, one expects that the level of safety will also increase and that the scarce resources available for fire safety will be utilized more efficiently.

After the Brown's Ferry Nuclear Plant fire in the USA some 20 years ago, there was a certain tendency in some plants to "throw money" at the problem of fire safety without always taking the time to ensure that the money was being thrown in the right direction. More recently, fire safety efforts have been much more focused, with detailed fire safety analyses (both FHAs and fire PSAs) helping to place attention on areas and systems in the plant which make a significant contribution to fire safety and to identify those areas where previous fire safety levels may have been overly conservative, if any.

3. FIRE SAFETY OBJECTIVES

To further complicate the situation, a number of different objectives must be considered for providing a specific level of fire safety in a nuclear power plant. Obviously, the primary objective is to ensure plant nuclear safety as defined in the IAEA's *Fire Protection Design Guide*, 50-SG-D2 [1]. The capability to achieve and maintain safe plant shutdown must be assured. Nonetheless, to overlook other issues related to fire safety would be a mistake.

These issues include limiting property damage as a result of fire and maintaining continuity of operations for the benefit of the public. That is, fire safety engineers should not be satisfied with a level of fire safety which allows a plant to be shut down safely during a fire emergency but remain in a shutdown state for 1–2 years or even indefinitely, due to extensive fire damage which was not "safety related." The minimum level of fire safety needed to ensure plant nuclear safety must be identified and provided—but design engineers should not stop at that point. We have an obligation to the global community to do a much better job than that.

4. MAINTAINING FIRE SAFETY LEVELS

Regardless of the defined design objectives or the analytical methodology used, once the required level of fire safety is identified, maintaining that level of safety is essential. One of the most effective means to accomplish this is to conduct periodic assessments to verify that the appropriate level of fire safety is being maintained in the plant. Analysis and identification of fire safety requirements are essential elements, but they represent only the first step in what must be a continuing process throughout the operational life of the plant. The periodic fire safety review process ensures that a consistent level of fire safety is maintained day after day and year after year—long after the initial glamour and attention of analytical studies has faded.

Nuclear plants expend significant time and valuable resources focusing on the analysis of low probability/high consequence events such as seismic, high wind, flooding and large fires, and rightly so. So it really is no surprise that fire safety is sometimes taken for granted in the day-to-day operation of the plant. It is all too easy to forget that small fires are a high probability/fairly high frequency event in nuclear plants—and small fires grow into large ones. For this reason, plant operators must continue to focus attention on maintaining the appropriate level of fire safety in each plant.

Maintaining fire safety in a complex industrial environment such as a nuclear power plant is a difficult task even where conditions are constant and unchanging. The difficulty is exacerbated by the fact that the nuclear power industry is not static. Individual nuclear power plants are very dynamic entities. Improvements in power plants seem to be a never-ending process. Improvements mean change; and changes result in modifications such as cable re-routing, penetrations to passive fire rated barriers, and increases in fire load. A continuing program of fire safety reviews and inspections will verify that design changes and plant modifications have been adequately assessed for fire safety impact. A comprehensive 10-year review of fire safety is appropriate and necessary, but it is not enough. A focused review of fire safety elements should be performed annually.

5. BENEFITS OF PERIODIC FIRE SAFETY REVIEWS

What benefits can be expected from performance of periodic fire safety reviews? These annual assessments will verify the operability of extinguishing systems whose function has been determined (through the FHA and/or fire PSA) to be critical to plant nuclear safety. The assessments will ensure that the combustible fire load does not exceed the level identified in the FHA. They will verify that the plant's fire prevention program is effective in controlling potential sources of ignition. They will also document the level of manpower, equipment and training provided by the fire brigade and assess the brigade's state of readiness for fire response.

Two examples will serve to illustrate the value of periodic fire safety reviews. The first example involves the case of a nuclear plant where it was determined that a fire extinguishing system was necessary to protect the bearings of the turbine generator. A carbon dioxide system was designed, installed, and maintained to satisfy the requirement. During a subsequent periodic fire safety review, the inspector noted that the carbon dioxide system was arranged to discharge its extinguishing gas outside the turbine shroud, rather than inside the enclosure where the majority of the oil hazard existed. Obviously, this fire protection system was totally ineffective; and the time, money and effort of installing the system was largely wasted. Had an oil fire involving the turbine bearings developed, the thermal detectors associated with the carbon dioxide system could have activated, uselessly discharging the system's gas into the large turbine building, while the lubricating oil continued to burn inside the turbine shroud and also spread onto the floor below via unprotected floor openings.

In another example, an interior standpipe system (or rising main) was installed in the reactor building of a nuclear plant for use by the fire brigade. During a periodic fire safety review, one of the fire hoses was chosen at random to verify the water flow available for fire brigade use. The control valve was opened, and an acceptable flow of water discharged from the nozzle for about 30 seconds—and then slowed to a bare trickle. Upon closer examination, it was discovered that the nozzle was totally blocked by clam shells. The plant obtained its water supply from an open water source (a river) which recently had become infested by Asiatic clams. These clams had entered the fire protection piping system and thrived. Fortunately, by discovering this problem during the periodic fire safety review, the plant was able to evaluate the situation, take appropriate action, and solve the problem prior to using the standpipe system in an emergency fire situation.

In addition to valuable troubleshooting, another immediate benefit of a fire safety review is the free and open exchange of fire safety knowledge and ideas between the fire safety assessor and the plant staff. Open communication should be encouraged at all levels of the plant. The intent of the review is definitely not to find fault nor to assign blame to a specific individual or group. Plant staff should be encouraged not to be defensive or worried that the assessor will identify a problem in their work area. An adversarial atmosphere is counterproductive. The atmosphere should be one which heightens fire safety awareness of plant staff and which promotes lively discussion on ways to improve fire safety. To be sure, one important objective of the fire safety review is to verify that the level of fire safety identified in design basis documents (such as the FHA, fire PSA, or the final safety analysis report) is being maintained. However, the overall intent of the review is to improve fire safety for the plant, not to take a narrow, legalistic approach which finds satisfaction in exposing deficiencies. The reviewer should make every effort to maximize the benefit to the plant, while minimizing any adverse effects on plant operations.

When problem areas are identified, they should be considered seriously and addressed promptly. A periodic fire safety assessment can identify small problems when they are not especially significant to nuclear safety, before they grow into crisis situations. On occasion, the management of a nuclear plant will mistakenly assume that the only correct resolution to fire safety problems is to establish a new working committee or administrative division, resulting in a virtual mountain of paperwork and documentation. Resolution of fire safety deficiencies may entail a certain amount of paperwork (after all, this is the nuclear power business). However, the intent of periodic fire safety reviews is not to create a paperwork nightmare; it is, quite simply, to improve fire safety.

Years of experience in performing fire safety reviews in nuclear power plants indicate that the initial one or two inspections at a plant will likely result in a number of findings. Some of these will be significant to plant nuclear safety, and many others will be of a minor nature. Subsequent inspections at the same plant usually show a marked reduction in findings (both in quantity and significance). Even so, such periodic reviews provide valuable assistance to the plant in continuing to improve the overall level of fire safety year after year.

6. FIRE SAFETY REVIEW PROCESS

The fire safety review should begin with a review of records and documentation. This documentation includes design basis documents such as the FHA and fire PSA, administrative procedures and policies related to the fire prevention program, and plant arrangement drawings. Discussion with plant staff at all operating levels is an important element throughout the review process. Maintaining open channels of communication is critical to allowing free exchange of ideas and information. Finally, an essential element of the review process is to conduct a field walkdown of accessible plant areas. The importance of this phase of the process cannot be overstated. Review of paperwork is necessary but should never be considered a substitute for visual observation in the field. Fully 30%–40% of the time allocated to the fire safety review should be spent in the field looking at actual plant conditions and talking directly with plant engineers, operators, maintenance staff, and fire brigade personnel.

It is important to recognize that effective fire safety reviews do not have to involve a major time commitment. A full-scale review of all elements of fire safety at a nuclear plant should require only about 100–200 manhours (for example, two engineers on site for two weeks each). An annual in-house review by qualified plant staff can be effective; however, occasional review by an independent, outside expert can bring in a fresh perspective. This independent review should be considered at intervals not exceeding 3–5 years.

The key to minimizing time and maximizing effectiveness of fire safety reviews is to choose personnel who are highly qualified and experienced. This is essential whether using in-plant staff or outside consultants. An option which provides a new, fresh perspective at a reasonable cost to the plant is to make arrangements with another nuclear plant to "trade" fire safety personnel for the purpose of conducting an independent review. This approach can be very effective in the exchange of new ideas and technology while at the same time meeting the need for an independent fire safety review at no additional cost to the plant.

7. SUMMARY

Effective fire safety requires the coordinated integration of many diverse elements. Clear fire safety objectives are defined by plant management and/or regulatory authorities. Extensive and time-consuming systematic analyses are performed. Fire safety features (both active and passive) are installed and maintained, and administrative programs are established and implemented to achieve the defined objectives. Personnel are rigorously trained. Given the time, effort and monetary resources expended to achieve a specific level of fire safety, conducting periodic assessments to verify that the specified level of fire safety has been achieved and is maintained is a matter of common sense. Periodic fire safety reviews and assessments play an essential role in assuring continual nuclear safety in the world's power plants.

REFERENCE

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Protection in Nuclear Power Plants: A Safety Guide, IAEA Safety Series No. 50-SG-D2 (Rev. 1), IAEA, Vienna (1992).

Invited Paper

FIRE PROTECTION ASSESSMENT IN A WANO PEER REVIEW

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Abstract

The peer review programme is becoming the key programme of WANO. The reviews are conducted to assess the performance of plant personnel, the conditions of systems and equipment, the quality of programmes and procedures, and the effectiveness of plant management. The review team consists of highly qualified staff from other WANO members throughout the world who have extensive practical experience in the area the review. At the request of Paris Centre Members, the fire protection area has been added to the scope of WANO peer reviews. Relevant performance objectives and criteria have been developed to cover this area, these are written guidances upon which review of plant performance can be based. They are supported by criteria, more narrow in scope, to help further define what attributes of the fire protection management area contribute to the achievement of the associated performance objective.

The driving force to establish WANO came from the accident at the Chernobyl Nuclear power plant in 1986.

This accident made nuclear operations aware of the need for international co-operation and the exchange of information.

WANO's mission is to maximise the safety and reliability of the operation of nuclear power plants by exchanging information and encouraging communication, comparison and emulation amongst its members.

Membership of WANO is open to all companies that operate electricity producing nuclear power plants and organisations representing nuclear operators.

Membership is entirely voluntary. Every single organisation in the world that operates a nuclear electricity generating power plant has chosen to be a member of WANO.

WANO is non profit making and has no commercial ties. It is not a regulatory body. It does not advise on design issues and it is neither a funding agency nor a lobbying organisation.

In short, WANO has no interests other than nuclear safety.

WANO operates five main programmes which form the basis of its activities:

- The Operating Information Exchange Programme
- The Operator to Operator Exchange Programme
- The Performance Indicator Programme
- The Good Practices Programme
- The Peer Review Programme

Each programme is designed to support the WANO mission and provide practical help to members.

I. THE PEER REVIEW PROGRAMME

The merit of periodically reviewing company operations by outside organisations or teams, already acknowledged in other industries, has been now developed by WANO within the nuclear electric power industry.

WANO peer reviews are voluntary and are performed at the request of a member: Utility or Plant.

The reviews are conducted to assess the performance of plant personnel, the conditions of systems and equipment, the quality of programmes and procedures and the effectiveness of plant management.

The focus is on plant safety and reliability.

The following areas are normally reviewed:

- Organisation and Administration
- Operations
- Maintenance
- Engineering Support
- Training and Qualification
- Radiological Protection
- Chemistry
- Operating Experience

The review team consists of highly qualified staff from other WANO members throughout the world who have extensive practical experience in the area they review. They bring together knowledge and experience of operating plants in different countries, and make an objective assessment of the operations of the plant reviewed against best international practice.
The review aims at identifying weaknesses, shortfall and their root causes. Suggestions to help the host plant in addressing and fixing problem can be made when enough expertise and knowledge are available among the team. It is then up to plant to make use of them or not.

Performance objectives and criteria have been developed to cover each area. These are written guidances upon which review of plant performance can be based. The performance objective states, in broad terms, what excellence in performance means for the specific management area. Excellence used in this context is not perfection, but is a dynamic performance goal that is always higher than the present level of performance.

The criteria are results-oriented. The methods for achieving the desired results are generally not stated. Thus considerable judgement is required in applying the criteria.

These performances objectives and criteria and the large international cross experience of WANO reviewers, serve as the review standard.

II . FIRE PROTECTION - REVIEW

Fire hazard is always considered as a significant risk in all industries. In a nuclear power plant this risk represents through its potential consequences a major threat to nuclear safety.

Therefore at the request of several members of the Paris Centre, this area was recently added to the scope of Paris Centre Peer Reviews.

A group of Experts from different utilities have contributed to set down the relevant performance objectives and criteria.

They will be used for the first time during two reviews which are under process at Doel and Hartlepool NPPs.
Fire Protection organisation, potential fire hazard, control of combustibles materials and ignition sources, Inspection, Maintenance and Test for both passive and active fire protection measures, Training and Knowledge will be reviewed.

After those two pilot Peer Reviews in the area of Fire Protection, the experience gained during this first implementation will be integrated as feedback in the next version of the PO and C.

III FIRE PROTECTION - PERFORMANCE OBJECTIVES AND CRITERIA

The Fire Protection area has been divided into the six following specific management areas:

FP.1 Fire Protection Organisation and Administration
FP.2 General Employee Knowledge in Fire Protection
FP.3 Fire Protection Surveillance, Testing and Maintenance Programme
FP.4 Fire Protection Work Practices
FP.5 Fire Protection Facilities and Equipment
FP.6 Fire Protection Personnel Knowledge and Performance

For each of those specific management areas, a performance objective is assigned.

Since, the performance objective is a broadly stated goal, achievement of the intent of the performance objective is desired. Therefore, criteria are assigned to each performance objective to help further define what attributes of that management area contribute to the achievement of the performance objective. The criteria that support each performance objective are more narrow in scope than the performance objective and typically describe a specific activity.

Since it is impossible to give a complete description of each performance objective and supporting criteria, one performance objective will be described. The concepts are similar among the other performance areas.

The following is the management area "Fire Protection Organisation and Administration", the performance objective and the associated criteria.

The performance objective states in broad terms:

"Fire Protection organisation and administration ensure effective implementation and control of fire protection activities".

Criteria associated with this performance objective state for example that the organisational structure should be clearly defined; staffing and resources should be adequate to accomplish assigned tasks, responsibilities and authorities should be clearly defined, personnel should understand their authorities and responsibilities, adequate training be insured.

Criteria also address management and supervisory involvement and attitude, contractors tasks responsibilities, authorities interfaces and co-operation between the various stakeholders.

However, the criteria listed are not intended to address every activity associated with the performance objective. Meeting all the listed criteria does not necessarily ensure that the performance objective is fully met. Conversely, a nuclear plant may effectively achieve the performance objective without meeting each specific criterion.

Therefore, it has to the emphasised that a plant should maintain a broad perspective and concentrate on achieving the intend of the performances objectives rather than focusing solely on the supporting criteria.

IV. FIRE PROTECTION - GUIDELINES

The Criteria associated with the Performance Objectives should not be considered as a check list.

The extensive practical experience, the knowledge and experience of all the peer review team members coming from different countries around the world will of course be used together to make an objective assessment against best international practices.

In addition, reviewers are asked to take advantage of publications of international bodies like:

- The Safety Guide of IAEA No. 50 SG D2, and

- The International Guidelines for the fire protection of nuclear power plants published on behalf of the national nuclear risks insurance pools and associations.

Lastly the reviewers are provided with guidelines for specific plant areas. Areas presenting a higher fire hazard or where the consequences of a fire are severe, need to be more deeply examined.

Here after are some examples:

1) In the main Control Room

Points that need particular attention:

- Fire protection equipment must not jeopardise the safety of the control room personnel (Carbon dioxide flooding system...)
- Fire hose reels and extinguishers close to the access
- Fire resistance rating (partitions, cables)
- Smoke dampers in ventilation systems, isolation of recirculation system
- Fire detection panel (presence of alarm, alarm sheets)

- Smoke detectors in the control room, cabinets and in the ventilation intake
- Breathing apparatus and protective clothes
- Combustible materials (carpeting, desk...).

The control room is generally the point of contact in case of fire. Therefore operators should be interviewed on:

- What are the criteria and means to call the internal and/or external fire brigade?
- What information must they communicate to the fire brigade?

The answers should be checked against the plant documentation and relevant procedures.

2) In the Electrical Room

A particular attention must be paid to the following rooms:

- Cable spreading room
- Plant computer rooms
- Switchgear rooms
- Safety-related battery rooms
- Galleries.

Some elements to observe are:

- In battery rooms: the ventilation (limit the hydrogen concentration and resistant to corrosive products), explosion proof lighting

- Fire compartment integrity (electrical and mechanical penetrations, gutters with hydraulic seal...)

- Posting and labelling (fire penetrations, fire doors, etc. well identified, emergency exits and escape routes clearly marked...)

- Phones in the vicinity

- Local detection cabinets and portable extinguishers located outside the protected area.

Conclusion

The Performance Objectives and Criteria provide a good support to define the scope of the review.

The guidelines are more like generic lists which must be completed to take into account specificity of the plant.

All this information is provided to the team members in advance of the review together with documentation relevant to the plant to be reviewed.

PERFORMANCE OBJECTIVES & CRITERIA

FIRE PROTECTION

FP.1 FIRE PROTECTION ORGANISATION AND ADMINISTRATION
FP.2 GENERAL EMPLOYEE KNOWLEDGE IN FIRE PROTECTION
FP.3 FIRE PROTECTION SURVEILLANCE, TESTING AND MAINTENANCE PROGRAM
FP.4 FIRE PROTECTION WORK PRACTICES
FP.5 FIRE PROTECTION FACILITIES AND EQUIPMENT
FP.6 FIRE PROTECTION PERSONNEL KNOWLEDGE AND PERFORMANCE

FP1

FIRE PROTECTION ORGANISATION AND ADMINISTRATION

PERFORMANCE OBJECTIVE
Fire protection organisation and administration ensures effective implementation and control of fire protection activities.

CRITERIA

A. The organisational structure is clearly defined.

B. Staffing and resources are sufficient to accomplish assigned tasks.

C. Responsibilities and authorities of all plant personnel involved in Fire Protection (including co-ordination of on-site and off-site fire fighting preparedness) are clearly defined and understood. Authorities are commensurate with responsibilities. Personnel are held accountable for carrying out assigned responsibilities.

D. Contractor tasks, responsibilities, authorities, and interfaces are clearly defined and understood.

E. Interfaces with insurers and official Safety Organisations are clearly defined and understood. Action items and recommendations receive appropriate priority and approval, and are scheduled and tracked to completion.

F. Interfaces with supporting groups, including headquarters organisations, are clearly defined and understood.

G. Managers and supervisors demonstrate and require a conservative approach concerning Fire Protection activities. High performance standards are established, communicated and reinforced. Managers and supervisors

H. routinely observe activities to ensure adherence to station policies and procedures, and to identify and correct problems.

I. Administrative controls are effectively implemented for Fire Protection activities.

J. Contractors and other non station personnel use the same (or equivalent) approved policies, procedures, control, and workmanship standards as station personnel.

K. Fire Protection problems and fire events are documented, evaluated and reported. These evaluations are reviewed for trends, and actions are taken to correct the causes.

L. Lessons learned from in-house fire event investigations and other industry operating experience are used to improve fire safety.

M. Personnel are actively encouraged to develop improved methods and a questioning attitude towards meeting safety, reliability and quality goals.

N. Fire Safety training programs are systematically evaluated and revised to ensure training is adequate and appropriate and that personnel are well trained.

O. Emergency plans for responding to fire are in place and are reviewed regularly for their *efficiency. The arrangements cover fire alone and fire occurring at the same time as a* nuclear accident

P. Modifications and design changes are reviewed appropriately to address the effects of the modification on fire safety. Staff assigned to undertake this activity are suitably qualified and experienced.

FP2

GENERAL EMPLOYEE KNOWLEDGE IN FIRE PROTECTION

PERFORMANCE OBJECTIVE
Station personnel, contractors and visitors have the knowledge necessary to implement fire protection practices associated with their work in an effective manner.

CRITERIA

A. Station personnel, contractors and visitors have a job-related fire protection knowledge and practical abilities, especially regarding the following :

1. actions to reduce ignition sources and fire hazards during routine operations.
2. action to minimise the accumulation of combustible materials.
3. actions to be taken in the event of a fire.
4. action to control and avoid the spreading of a fire

B. Job-related knowledge and practical ability are maintained in the following areas:

1. Basic fire protection subjects
2. Pertinent changes in fire protection procedures
3. Lessons learned from in-house and industry operating experiences

C. Personnel are aware of the fire protection requirements for the jobs they perform. Hot work permit procedures and their requirements are implemented on the job performance. Proficiency is demonstrated in using extinguishers, fire hose stations or installed fire fighting equipment.

FIRE PROTECTION SURVEILLANCE, TESTING AND MAINTENANCE PROGRAM

PERFORMANCE OBJECTIVE

Surveillance, testing and maintenance program ensure optimum performance and reliability.

CRITERIA

A. Codes, regulations and standards applicable to the fire protection systems and equipment are clearly understood.

B. All spurious fire alarms are reported and investigated. Deficiencies are corrected as soon as possible.

C. A comprehensive surveillance, maintenance and testing programme is established and implemented to cover both active and passive fire protection features of the plant.

D. The frequency and scope of inspection, maintenance and testing activities are appropriate to the individual parts of the fire protection system and are in accordance with best international practices.

E. Inspection, maintenance and testing activities are carried out by suitable qualified and experienced staff.

FP 4

FIRE PROTECTION WORK PRACTICES

PERFORMANCE OBJECTIVE

Station fire protection work practices and conditions achieve a high degree of safety

CRITERIA

A. The use and storage of combustible materials (solids, liquids, gases) is minimised. Accumulations of transient combustible materials and wastes are controlled. Safe practices are used in the storage, handling, use and transportation of combustible substances. Storage areas are routinely monitored and precautions are taken to limit the potential consequences of a leakage (for example use of drip trays for combustible liquids and sufficiently ventilated areas for flammable gases).

B. The use of ignition sources associated with hot work processes (cutting, welding, grinding, hot roofing work, etc.) is adequately controlled to prevent fire from starting.

C. Fire protection systems are maintained operable and reliable to the maximum extent possible. Controls are established to ensure plant safety is maintained. Compensatory measures, are implemented when fire protection systems are found to be defective or placed out of service.

FIRE PROTECTION FACILITIES AND EQUIPMENTS

PERFORMANCE OBJECTIVE

Fire Protection facilities and equipment of appropriate capability and capacity reduce the probability and consequences of fires to a minimum.

CRITERIA

A. A comprehensive evaluation of fire hazards is available for the plant and kept up to date. The scope of the evaluation covers the threat of fire to personnel, to nuclear safety and to the operation of the plant

B. The passive fire protection features are well defined, well identified and maintained in order to avoid the fire spread. The plant is subdivided into individual fire compartments and fire cells to reduce risk of spread of a fire and to prevent common mode failure of redundant nuclear safety related systems.

C. An adequate fire detection and alarm system is operational and efficient.

D. An adequate gas detection and alarm system is operational and efficient.

E. Fixed fire extinguishing systems are appropriate for the hazard they protect, readily identified, and are in operational service. Their failure, rupture or inadvertent operation does not impair the operation of safety systems required to a safe shutdown of the plant.

F. Portable fire extinguishers of appropriate types, fire hydrants and hoses reels are suitable located and provided in a sufficient number to ensure efficient and rapid manual fire fighting according to the size and the nature of the fire load.

G. The fire fighting team is provided with sufficient mobile equipment to allow fires to be fought in all parts of the plant.

H. Emergency lighting and communication devices are operational and efficient. Escape routes and access routes for fire fighting are clearly marked and free of obstruction.

I. Suitable and sufficient personnel protective equipment is provided for the fire fighting team. Storage areas are easily accessible and well known by the fighting team.

J. **NTOL:** During construction stage, a provisional system of fire hydrants is made available as soon as possible and emergency protection in the form of fire extinguishers and hose lines is provided.

K. **NTOL:** In case nuclear fuel elements have to be stored on site before the facilities in the fuel building are ready, the temporary storage facility should be protected against fire and the storage arrangement should be such that water used for fire fighting cannot lead to criticality conditions.

FIRE PROTECTION PERSONNEL KNOWLEDGE AND PERFORMANCE

PERFORMANCE OBJECTIVE

Fire protection personnel knowledge, training, qualification and performance support effective implementation of fire protection and fire fighting practices.

CRITERIA

A. Fire protection activities are performed by or under the supervision of personnel who are qualified for the tasks they perform.

B. The fire safety training programme is documented and includes :

practical training :

1. practical training and exercises on fire fighting techniques including the use of breathing apparatus.
2. The actions of the fire team in the event of a fire including muster, deployment, command, control and liaison with off-site brigades.

theoretical training :

1. general information and functions such as : plant lay out and emergency evacuation routes, reporting relationship, communication methods, document and procedure issue and revision, record management, material procurement and industrial safety practices
2. plant components and system fundamentals including nuclear and fire related hazards.
3. performance of fire prevention surveys, including data collection, analysis and documentation and the selection, inspection, use and care of appropriate fire fighting equipment.
4. fire protection theory and techniques (fire safety culture) including fire protection standards, regulations, work control and other job responsibilities.
5. plant specific application of appropriate lessons learned from in-house and industry operating experience.

C. On the job training requirements are identified, completed and documented prior to assignment to perform task independently.

D. Continuing training addresses areas that include hardware and procedure changes, and infrequently used skills.

E. Fire protection personnel are capable of diagnosing and initiating corrective actions for unusual conditions during routine and accident situations

F. The knowledge and practical abilities of contract fire protection technicians are equivalent to those of station fire protection personnel for the function to which they are assigned on the station

G. **NTOL:** Experiences by using fire protection equipment's similar to those found in power plants are provided to inexperienced personnel assigned to fire protection functions.

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Invited Paper

SOME INSIGHTS FROM FIRE RISK ANALYSIS OF US NUCLEAR POWER PLANTS

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Abstract

Fire risk analysis has been conducted for a significant portion of the nuclear power plants in the U.S. using either Probabilistic Risk Assessments (PRAs) or FIVE or a combination of the two methodologies. Practically all fire risk studies have used step-wise, screening approach. To establish the contents of a compartment, the cable routing information collected for Appendix R compliance have been used in practically all risk studies. In several cases, the analysts have gone beyond the Appendix R and have obtained the routing of additional cables. For fire impact analysis typically an existing PRA model is used. For fire frequencies, typically, a generic data base is used. Fire scenarios are identified in varying levels of detail. The most common approach, in the early stages of screening, is based on the assumption that given a fire, the entire contents of the compartment are lost. Less conservative scenarios are introduced at later stages of the analysis which may include fire propagation patterns, fires localized to an item, and suppression of the fire before critical damage. For fire propagation and damage analysis, a large number of studies have used FIVE and many have used COMPBRN. For detection and suppression analysis, the generic suppression system unavailabilities given in FIVE have been used. The total core damage frequencies typically range between $1x10^{-6}$ to $1x10^{-4}$ per year. Control rooms and cable spreading rooms are the two most common areas found to be significant contributors to fire risk. Other areas are mainly from the Auxiliary Building (in the case of PWRs) and Reactor Building (in the case of BWRs). Only in one case, the main contributor to fire is the turbine building, which included several safety related equipment and cables.

1. INTRODUCTION

Fire risk analysis has been conducted for a large number of the nuclear power plants in the U.S. Since 1980, several utilities have conducted Probabilistic Risk Assessments (PRAs) that have addressed the contribution of internal fires to plant risk. In addition, in the last few years, as part of their compliance efforts for the Independent Plant Examination for External Events (IPEEE) requirements [1], most utilities have elected to use a fire risk analysis method to address the fire vulnerabilities in their plants.

The focus of this article is the insights gained from a large number of fire risk studies. The phrase "fire risk studies" used in this article stands for those studies that the authors have either reviewed or had direct involvement in their preparation. The insights are presented in terms of key topics that the authors deem as important to fire risk analysis.

2. SELECTED METHODOLOGY

Early risk studies were based primarily on fire PRA [2, 3]. Recent studies are based on either PRA, FIVE [4], or a hybrid of these two methodologies. FIVE was developed to provide the utilities with a simplified methodology to be used in complying with the IPEEE requirements. As it is stated in Ref. [4], FIVE was developed to support a PRA. From a review FIVE it can be concluded that fundamentally there are many similarities between the FIVE and PRA methodologies, especially in the context of the screening analyses, which is a critical step for a robust and complete fire risk assessment.

Many fire risk studies have used a hybrid of FIVE and PRA methodologies. Common areas in which the FIVE methodology was altered, either towards a more or less detailed analysis, include the analyses of fire detection and suppression timing, fire compartment interaction, manual fire fighting, plant recovery, and data input for fire growth/propagation. At the same time, typical PRA analyses also omitted some aspects of a state-of-the-art PRA. Typical areas of omission or simplification in the PRA-based analyses included: treatment of fire growth, propagation and damage, detection and suppression, and manual fire fighting; simplified plant impact modeling; limited modeling of remote shutdown scenarios; generic plant recovery modeling; and limited treatment of control circuit failure modes, human factors and recovery actions.

3. FIRE IGNITION FREQUENCY

Most fire risk studies have used generic fire occurrence frequencies without updating them with plant specific experience. The data provided in Ref. [4] is used extensively in the recent studies. Almost all fire risk studies have adjusted the overall fire occurrence frequencies to establish the fire frequency for individual plant locations. In a few cases the frequency is further adjusted to account for the severity of the fire.

Several studies have screened out compartments based solely on fire ignition frequency. The sole content of these compartments is typically cables qualified per IEEE fire ignition standards. For example, for a cable chase area, it is argued that since all the cables are qualified per IEEE standards, the area is not visited often and there are no other equipment, the fire frequency is small enough that the compartment can be screened from further analysis. This conclusion is reached without a review of potential equipment and instrumentation damage possibilities, impact on the plant as a whole and especially the operators' response to the potential instrumentation and control circuit failures. Given the large uncertainties in fire occurrence frequencies for such compartments, an early screening practice does not allow for plant personnel to gain a clear appreciation of potential accident sequences.

4. CABLE ROUTING INFORMATION

Cable routing information is perhaps the most important element of a nuclear power plant fire risk study and the routing of a select set (a large number) of cables is necessary for this purpose. Errors in this aspect of the evaluation can jeopardize the validity of the entire analysis. This information, is often available in the form of computerized databases, and for some plants, the retrieval of this information may prove to be very time consuming. The cable routing information, almost invariably for all risk studies of U.S. plants was taken from the data established as part of the fire hazard analysis conducted for compliance with U.S.NRC fire protection requirements. (We will refer to these as Appendix R requirements.) Appendix R requires a safe shutdown analysis that results in the identification of equipment, associated circuits and cables that are needed to ensure safe shutdown given a fire in a specific compartment.

There are some differences between a PRA model and the safe shutdown model, which lie primarily with modeling of the containment functions and initiating events. Also, in practice, we have found differences in modeling assumptions which have led to differences in selected cables and circuits. The amount of additional information (that is cable routing beyond that provided through the Appendix R effort) incorporated into the analysis varies significantly among the fire risk studies. A significant portion of the studies did not seek additional information, and therefore, had to make conservative assumptions or pass judgment in the location of certain cables (a source uncertainty).

5. EQUIPMENT FAILURE MODES AND INITIATING EVENTS CAUSED BY A FIRE

The possibility of a fire causing an initiating event other than reactor trip has been explicitly addressed in most fire risk studies. However, in many cases the extent of the treatment has been limited. For example, the possibilities of spurious opening of pressure operated relief valve (PORV), or loss of offsite power, have not been treated in some PWR studies. Often the safe shutdown analysis conducted for compliance with Appendix R requirements does not address all those initiating events that are considered in an internal events PRA. Related to initiating events, the failure modes of control and instrumentation circuits must be investigated and often in great detail. A cable failure under fire conditions may cause a combination of shorts among various wires of a circuit. One set of the shorts may lead to spurious actuation of equipment or damage to equipment in such a way that further recovery of the failure may not be possible. Probabilistic arguments have been used to screen these failure modes. The probabilities are based purely on judgment and are not supported by any field observations.

6. THRESHOLD VALUES FOR SCREENING

All fire risk studies have included at least one screening step to reduce the number of compartments and fire scenarios requiring detailed analysis. Typically a large fraction of the compartments are screened out and a small number are retained for detailed analysis. Qualitative or quantitative methodologies have been used for this purpose. In both cases it is assumed that given a fire, the entire contents of the compartment is failed. The qualitative method is typically based on the presence of safe shutdown cables and equipment in a compartment. The most common quantitative methodology is based on core damage frequency. If the frequency is above a threshold value, the corresponding fire scenario or compartment is subjected to detailed analysis. The threshold value typically employed by a large number of studies and recommended by FIVE, is 10^{-6}/ry. A minority of risk studies have used 10^{-7}/ry and 10^{-8}/ry for this purpose. The benefits of using a low threshold value, because of rare application of such values, could not be readily assessed because these studies were also accompanied by non-conservative assumptions or methodology practices.

7. FIRE PROPAGATION AND DAMAGE MODELING

Fire risk studies completed prior to the publication of FIVE have either used COMPBRN [5] or have made conservative assumptions to avoid the use of fire growth models. For example, it was assumed that fire would damage all components within a given compartment where the fire originated, and did not model any other possibility. In other studies, it was assumed that fire suppression would be effective without consideration of the relative timing of damage and suppression effectiveness. The risk studies completed in the past few years have utilized FIVE look-up tables extensively. Both the FIVE tables and COMBPRN code have to be used with caution, otherwise physically unrealistic fire damage scenarios may result. This is especially true with regard to fires that might spread beyond the ignition source.

A number of fire risk studies were found to include a variety of optimistic or otherwise inappropriate assumptions. For example, inadequate consideration has been given to transient combustible and fixed combustible fire sources, cable qualifications have been used as the basis for screening compartments, and temperature damage thresholds and heat loss factors (e.g., a value of 0.85 has been used in place of 0.7) have been inappropriately modeled. These factors may have likely led to inappropriate screening of fire areas, underestimation of core damage frequency contributions from fire areas that survived screening, and (in some cases) the overestimation of the risk significance of non-critical fire areas (thus masking those areas most critical to fire risk).

Fires originating in electrical cabinets (of all sizes and service types) are found to be important to risk, in part due to the co-location of these cabinets with electrical cables. Plant-specific details of electrical cabinets are found to be important. At one plant, penetrations where cables exited the top of the switchgear cabinets were not adequately sealed, which provided an exit pathway for the chimney effect. This situation allowed for the fire to be postulated as propagating up and out of the switchgear cabinet. At another plant, control cables in the control room were arrayed across the top of control room cabinets with open tops. Again, this led the analysts to postulate cabinet fires that propagate to the overhead cables.

The heat release rate and the potential for propagating to an adjacent cubicle are two important factors of cabinet fire modeling. Sandia test results [6] provide a basis for these factors. However, large variations exist in the interpretation of the test results, which has led to optimistic assumption in some of the fire risk studies. Modeling of electrical cabinet fires in control rooms is also very important since such fires can force abandonment of the control room due to smoke accumulation. This issue is further discussed below.

8. ANALYSIS OF FIRE DETECTION AND SUPPRESSION

There is a large variation among the fire risk studies in the method for modeling fire detection and suppression. Many fire risk studies did not model manual suppression of fires (except in the case of control room fires). This is supported by the fact that fire data indicates that in most situations fire brigades successfully extinguish or control fires within 30 minutes [7] and that fire models often predict damage to critical components in much less than 30 minutes. Not modeling fire brigade actions is functionally equivalent to assuming that there is a negligibly low conditional probability that the brigade will unintentionally damage equipment which has not been damaged by a fire. Of course, for those studies which have assumed that fires would damage all components within the compartment, secondary damage due to the fire brigade suppression activities is implicitly included.

The failure probability of the suppression system is often gleaned from either FIVE or other industry sources. Many fire risk studies have multiplied this failure probability with the fire occurrence frequency. This makes the assumption that a fire damages the entire contents of the compartment and the suppression system, if it functions properly can prevent all damage. This may represent an optimistic assumption if the layout of cables and equipment within the compartment is not examined. If a critical set of cables and equipment are within a small area inside the compartment, and especially on top of a likely ignition source, this multiplication process is certainly optimistic. Several studies have used generic values for suppression system reliability. This, too, may be optimistic if the fire detection and suppression systems of a plant are not compliant with the fire codes.

Interaction between automatic fire suppression systems and safety related equipment is rarely considered. For example, CO2 actuation circuit, in one case, was interlocked with diesel generator control circuits and could potentially prevent emergency generator start-up.

9. INTER-COMPARTMENT FIRE PROPAGATION

The possibility of inter-compartmental fire propagation has the potential for causing damage to cables and equipment of multiple safety trains. This aspect of fire analysis has been treated in the risk studies with varying degrees of detail and sophistication. In a large number of studies the Fire Compartment Interaction Analysis (FCIA) of FIVE has been employed. In these studies passive fire barriers are assumed to be 100% reliable. Also, this assumption is often extended to active fire barriers.

An important element of this issue is the method by which compartments are defined. Aside from the upper floors of a typical Reactor Building in a BWR, a large majority of the compartments in nuclear power plants are defined by fire barriers that are rated to contain the effects of a fire for one to three hours. However, practically all the compartments have passages to adjacent compartments via doors ventilation ducts, etc. An important mechanism for the propagation of the effects of a fire is the escape of hot gas layer through these openings. A large number of risk studies have ignored the following three potential fire propagation scenarios:

- Failure of an active fire barrier (e.g., self closing doors and fire dampers) to close, and propagation of hot gases to adjacent compartments.
- Failure of fire barrier integrity due to fire-fighting activities (e.g., opening of doors to gain access or route hoses).
- Failure of a fire barrier from being overwhelmed by an excessive source (e.g., diesel fuel tank fire, or the walls separating the turbine building from the rest of the plant).

In some cases, the possibility exists for the fire-fighting activity to lead to a breach in the integrity of the fire barriers. For example, if trains A and B of a safety system are located in two adjacent rooms that are connected by a door, the possibility exists for the fire brigade personnel to enter the affected room through the unaffected one, and if the door is left open and the fire continues to burn, it is possible for the effects of the fire to propagate through the open door.

10. CONTROL ROOM FIRE MODELING

A significant number of fire risk studies have found the control room as one of the most important fire risk contributors. Other studies have identified very low control room fire core damage frequencies and have screened out the control room as a fire-induced core damage contributor. The differences lie primarily in modeling assumptions than in the physical layout of the control room or training of the operators. Some studies have used a detailed analysis of the fire incidents in the control room and many have used a simplistic approach where a conditional probability is assigned to the failure of controlling the plant from outside the control room. In the simplistic approach no further evaluation of the possible operator action scenarios are attempted. In the detailed approach, every cabinet section is examined for potential fire ignition, failure in the control circuits, and operator response to the specific set of failures in addition to the fire detection and fire fighting activities. In these studies, the remote shutdown panel (in some plants this may include several separately located panels) is analyzed for potential failures from control room circuit damages and for operator errors in its proper usage.

Ease of fire detection and suppression is the main reason cited by those studies that have concluded low control room fire risk. Control room fire non-suppression conditional probabilities in the range of 1-3% have been used as compared to more typical values of 2-5% used for suppression systems in other parts of the plants. Some studies specifically cited a 15-minute time period before control room abandonment would be required. This value assumes in-cabinet smoke detectors are present. With only general area detectors, the proper interpretation of the Sandia-sponsored tests yields an estimate of seven minutes. A few studies assumed that smoke-forced abandonment of the control room would occur only if multiple cabinets were involved in the fire, which is also inconsistent with Sandia test results [6].

In addition to the above, it may be noted that in at least two cases, the control room is shared between two units. Fire damage in the cabinets in one unit can force the abandonment of the control room for the other unaffected unit as well. The human error coupling between the two units is not addressed in any of the studies. Also, most fire risk studies have not used a systematic method to verify control systems interactions. Typically, circuit isolation capability, remote location, and procedures are used to insure that there would be no adverse interactions between the control room and remote shutdown panel.

11. HUMAN ACTIONS AND FIRE RISK

Human errors, typically those involving recovery actions, is found to be important in most fire risk studies. It is not clear, however, that human performance has been adequately assessed. Many studies have used internal events human reliability models and data without accounting for the unique aspects of fires. For recovery actions which take place in the area where the fire occurs, or which require operators to pass through the area where the fire occurs, it is inappropriate to use the internal events recovery probabilities directly without providing justification. Use of internal events human reliability models for external event-initiated scenarios, such as fires, must be approached cautiously due to the differences in performance shaping factors (PSFs) between a fire incident and otherwise. Practically none of the fire studies have addressed the possibility of wrong information on the control board and errors of commission as a results of that.

Human error may occur during fire fighting. However, nearly all risk studies have not included the potential for adverse effects of manual fire-fighting efforts on safe shutdown equipment. An error by members of the fire brigade (e.g., misdirected water stream) may fail equipment other than those exposed to the fire.

12. SMOKE CONTROL

Only a handful of fire risk studies have addressed the possibility of smoke propagation, none have considered the possibility of short term effects of smoke on equipment and a few have considered, albeit qualitatively, the possibility of suppression system activation from smoke migration in compartments other than the fire origin. In the latter case, possibility of equipment damage from exposure to fire suppression medium may be of concern. Almost all risk studies have not included the potential for smoke to hinder manual fire-fighting effectiveness or misdirect suppression efforts. It must be noted that FIVE [4] specifically states that degradation of equipment from secondary (non-thermal) fire environmental effects can be ignored.

13. CORE DAMAGE FREQUENCY

Core damage frequency is used in almost all fire risk studies as a measure to screen-out, or to establish the importance level of various fire compartments. The total core damage frequency for fire events spans a large range, but the majority of the risk studies report a total core damage frequency between 10^{-6} and 10^{-4} per reactor year. Based on these results one can conclude the following:

- Fire remains a significant contributor to core damage frequency for a large fraction of the plants.
- There is a significant plant-to-plant variation in terms of the reported overall fire risk, especially when the specific contributors are considered. Most of this variability can be attributed to differences in methods and assumptions employed in the analysis.

Fire is a significant core damage contributor because its occurrence rate is comparable to many internal initiating events, and in addition, it can simultaneously disable several pieces of equipment. Furthermore, fire may influence operator performance and thus further increase the probability of failure to recover from the fire event.

The plant-to-plant variation of risk among the plants can be attributed to two issues—variation in plant layouts, and variations in underlying assumptions and application of fire risk methodologies. The underlying assumptions refer to such issues as considering the cable spreading room free of transient fuels, screening out cable chases that include a large number of cables from various systems/trains, the wide variation in methods and parameters used for room screening, and the wide variation in using optimistic parameter values for fire propagation or suppression modeling.

14. DOMINANT RISK CONTRIBUTORS

The dominant risk contributors are presented in terms of two important aspects of a fire event: (1) location of the fire, and (2) equipment/systems affected by the fire. It is very common for the control room and cable spreading room to be reported as two of the most significant fire risk contributors. Although there are no clear patterns among the plants, and one cannot draw any clear conclusions without full knowledge of the specific cables present in those rooms identified as significant, it can be stated that buildings which house non-safety related equipment and cables, with the exception of the turbine building, have been found to be of little or no fire-risk significance.

For both PWRs and BWRs, the control room is the most often quoted significant fire-risk contributor. The dominant sequence is typically a fire in a vital control panel that leads to control room evacuation and failure of the operators to successfully use the alternate shutdown panels. The cable spreading and high-voltage switchgear rooms have been mentioned in approximately one-half of the risk studies as a dominant fire risk contributors. Since cable spreading rooms contain almost the same set of control and instrumentation circuits as those in the control room, one would expect the cable spreading rooms to be identified as risk significant for as many cases as where the control room was identified as a dominant risk contributor. Differences can be attributed to the fact that several risk studies have screened out the cable spreading room by either concluding that fire scenario frequencies in the cable spreading room are too small, or by using qualitative arguments regarding fire ignition possibility in the room. Only in a small number of plants there are multiple and well separated cable spreading rooms serving the control room. Typically, in such cases, a cable spreading room may only contain one train of the vital circuits, and thus, the fire events in those rooms are found to be insignificant contributors. Operator actions are a key element of fire scenarios associated with control room or cable spreading room fires.

The service water and component cooling-related areas have also been reported in several risk studies as being important fire risk contributors. Various auxiliary/reactor building and turbine building areas have been included in these lists as well. However, in the case of the latter two buildings, there is no overall pattern. This observation is expected, since generally, there are no common patterns among the auxiliary and reactor buildings across the plants licensed in the U.S. In only a few cases was fire affecting the entire turbine building found to be significant. In most turbine building assessments, the dominant fire scenarios are attributed to a compartment or a localized area that is part of the turbine building.

Regarding dominant accident sequences, the level of information varies considerably among the fire risk studies. Licensees have almost in all cases used an existing internal events model (i.e., event trees and fault trees) to determine the core damage frequency contributions of various fire scenarios. Often only a portion of the internal-events model has been utilized—either because of a lack of sufficient information on cable routing, or to simplify the core damage analysis. In one case, only one sequence, from the multiple number of sequences, was used to model the fire impact on plant safety. In a few cases, the possibility of a LOCA from a fire has been considered explicitly. In most cases, it is concluded that a fire-induced LOCA is not possible. In the case of PWRs, a large majority of the studies have taken credit for the possibility of feed-and-bleed.

15. CONCLUSIONS

A large number of fire risk studies have been completed for the nuclear power plants in the U.S.A. A number of insights can be noted from a review of these studies. The most important is that fire is an important risk contributor to plant risk. Other insights can be summarizes as follows:

- Either PRA or FIVE or a combination of the two methodologies have been used. In many cases the analysts have modified FIVE procedures to match their specific needs.
- For most plants, the critical fire areas include the control room, cable spreading room, and electrical rooms.
- Fire core damage frequencies span a wide range -- from 10^{-6} to 10^{-4} per reactor year.
- In none of the fire risk studies have multi-compartment fire scenarios been found to be an important risk contributor. This is in part based on the assumption made in many fire risk studies that active fire barriers are highly reliable.
- In almost all risk studies operator actions are critical to the reduction of fire risk.
- None of the fire scenarios identified in the risk studies were found to fail a minimal cutset of equipment leading to core damage. In other words, additional failures, somewhat independent of the fire, have to occur for core damage to be realized. This conclusion confirms the objectives of NRC fire protection requirements (i.e., Appendix R).
- Generic values have been used for suppression system reliability.
- The possibility of barrier failure because of large quantities of combustible materials concentrated in one area has not been considered.
- Several studies have screened out compartments that contain an important combination of safety related systems based solely on the frequency of fire occurrence. For example, for a cable chase area, it is argued that since all the cables are qualified per IEEE standards, the area is not visited often and there are no other equipment, the fire frequency is small enough that the compartment can be screened from further analysis. This conclusion is reached without a review of potential equipment and instrumentation damage possibilities, impact on the plant as a whole and especially the operators' response to the potential failures. Given the large uncertainties in fire occurrence frequencies for such compartments, an early screening practice does not allow for the plant owner to gain a clear appreciation of potential accident sequences.
- Operator actions in response to the effects of fire on systems are rarely modeled in detail. Several studies appear to have applied values directly taken from internal events analysis without correcting those values using performance shaping factors influenced by the effects of fire.
- It is difficult to draw meaningful comparisons, with regard to trends in fire risk, among reactor types (i.e., PWR and BWR), and among the architect-engineers because of the variation in assumptions, methods, and input data among the fire risk studies.

It is interesting to also note that very few of the risk studies have directly led to plant improvements. Most improvements specifically attributed to a fire risk study are in the area of procedural enhancements. Modeling of human performance in fire scenarios must be generally regarded as a weak area. The lack of details regarding human performance modeling precludes significant reliance on human performance insights from the fire risk studies.

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Invited Paper

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OVERVIEW OF IAEA GUIDELINES FOR FIRE SAFETY INSPECTION AND OPERATION IN NUCLEAR POWER PLANTS

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Abstract

In 1992, the International Atomic Energy Agency began an ambitious project on fire safety in nuclear power plants. The purpose of this ongoing project is to provide specific guidance on compliance with the requirements set forth through the IAEA Nuclear Safety Standards program established in 1974. The scope of the Fire Safety project encompasses several tasks, including the development of new standards and guidelines to assist Member States in assessing the level of fire safety in existing plants. Five new Safety Practices, one new Safety Guide and a Technical Document have been developed for use by the fire safety community. The primary intent of these new documents is to provide detailed guidance and a consistent format for the assessment of the overall level of fire safety being provided in existing nuclear power plants around the world and especially in developing countries. Sufficient detail is provided in the Safety Guide and Safety Practices to allow technically knowledgeable plant personnel, outside consultants or other technical experts to assess the adequacy of fire safety within the plant facilities. This paper describes topics addressed by each of the IAEA Fire Safety documents and discusses the relationship of each document to others in the series.

1. INTRODUCTION

In 1992, the International Atomic Energy Agency (IAEA or the Agency) began an ambitious project on fire safety in nuclear power plants. The purpose of this ongoing project is to provide specific guidance on compliance with the requirements set forth through the IAEA Nuclear Safety Standards (NUSS) program established in 1974. Previous safety review documents and services offered as part of this program were not sufficiently detailed to address the full range of fire safety issues in existing nuclear power plants to a level which could provide significant assistance to the Member States in improving their fire safety programs. The scope of this new project encompasses such issues as developing new standards and guidelines to assist Member States in assessing the level of fire safety in existing plants; conducting fire safety review missions for specific facilities; presenting interregional fire safety training courses; and designing a program to initiate the collection of fire related statistical data of value to existing nuclear power plants, both from an equipment reliability standpoint and for fire safety lessons learned.

Over the course of the last five years the Agency has made significant progress towards the goals of its Fire Safety project. Five new Safety Practices, one new Safety Guide and a Technical Document have been developed for use by the fire safety community. Six fire safety review missions have been conducted at nuclear power plants, and two fire safety training courses have been organized and implemented by the Agency. This discussion focuses on the publications developed as a result of the IAEA Fire Safety project.

Beginning in the spring of 1992 and continuing through the fall of 1996, initial draft documents for the seven new publications were created. Formal review and comments on each document were solicited from the technical representatives of the Member States. Technical enhancements were incorporated during the review process through a series of technical meetings held at the Agency's offices in Vienna. The result of this project is a formal series of guidelines for fire safety inspection and operation of existing nuclear power plants which have been published by IAEA (or are currently in process) and which are available for use by the Member States.

The primary intent of these new documents is to provide detailed guidance and a consistent format for the assessment of the overall level of fire safety being provided in existing nuclear power plants around the world and especially in developing countries. Sufficient detail is provided in the Safety Guide and Safety Practices to allow technically knowledgeable plant personnel to assess the adequacy of fire safety within their own facilities. The documents also can be used by outside consultants or other technical experts to assist the plant staff, when appropriate. The titles, document numbers and publication dates of the seven new IAEA fire safety documents are as follows:

(a) *Fire Safety During Operation of Nuclear Power Plants*—Safety Guide 50-SG-xx (in review process) [1]
(b) *Inspection of Fire Protection Measures and Fire Fighting Capability at Nuclear Power Plants*—Safety Practice 50-P-6 (1994) [2]
(c) *Assessment of the Overall Fire Safety Arrangements at Nuclear Power Plants*—Safety Practice 50-P-11 (1996) [3]
(d) *Evaluation of Fire Hazard Analyses for Nuclear Power Plants*—Safety Practice 50-P-9 (1995) [4]
(e) *Preparation of Fire Hazard Analyses for Nuclear Power Plants*—Safety Practice 50-P-xx (in publication process) [5]
(f) *Treatment of Internal Fires in Probabilistic Safety Assessment for Nuclear Power Plants*—Safety Practice 50-P-xx (in review process) [6]
(g) *Guidelines for IAEA Fire Safety Review Services at Nuclear Power Plants*—TECDOC-xxx (in publication process) [7]

2. FIRE SAFETY GUIDES

IAEA Safety Guides have been developed to assist Member States in the implementation of the requirements set forth in the NUSS Codes, which are intended to offer a basis for nuclear power regulation within the Member States. The Safety Guides suggest methods of implementation for parts of each Code, and these methods are further elaborated within the Safety Practices. The NUSS Codes, Safety Guides, Safety Practices and other Technical Documents are intended to be used in conjunction with one another and contain considerable cross-referencing to one another, along with a list of all NUSS titles.

2.1. *Fire Protection in Nuclear Power Plants*

Just prior to the initiation of the current IAEA Fire Safety project, IAEA published a major revision to its first Safety Guide on fire protection. Safety Guide 50-SG-D2 (Rev. 1), entitled, *Fire Protection in Nuclear Power Plants* [8], provides detailed fire safety requirements for use by designers, regulators and safety assessors. Although this Guide does contain some information pertaining to fire safety during the operation of nuclear power plants, its primary focus is on incorporating fire safety measures into the design and construction of new plants. Both this Safety Guide and the other documents discussed in this paper are intended to apply to "land based nuclear power plants with thermal neutron reactors of general use such as light water, heavy water or gas cooled types" [8].

This design Safety Guide laid the foundation upon which the other Fire Safety documents have been built, particularly regarding definitions for such terms as *design basis accidents, nuclear safety, safety systems, fire, fire compartment, fire cell, redundant equipment,* and *single failure criteria.* Subsequent documents also incorporate this Guide's definition of the overall fire safety objectives needed to ensure plant nuclear safety (e.g., prevention, detection/extinguishment, and containment), the degree of redundancy required for safety systems, and the *defense-in-depth* concept as applied to nuclear power plants.

2.2. *Fire Safety During Operation of Nuclear Power Plants*

The draft Safety Guide 50-SG-xx, entitled, *Fire Safety During Operation of Nuclear Power Plants,* applies the principles set forth in the design Safety Guide to the management and operation of a nuclear power plant. The new Safety Guide is intended for use by plant managers and operators, as well as regulators and safety assessors. It elaborates upon operational topics introduced in the design Safety Guide, such as fire prevention, control of combustibles and ignition sources, safety culture, manual fire fighting, training, and quality assurance. The new Guide differs from its predecessor by discussing additional operational topics, such as organization and responsibilities; periodic updating of the fire hazard analysis; design changes and plant modifications; inspection, maintenance, and testing of fire protection measures; and records and documentation. The Guide even includes an appendix which contains a sample list of those features, systems, equipment, and components that should be included in a fire safety inspection, routine maintenance, and testing program. The objectives of this publication are to ensure that the original fire protection design intent is maintained, to ensure that design changes and plant modifications are addressed in a timely manner, to ensure that effective fire prevention measures are implemented throughout the operating life of the plant, and to develop and maintain a consistent fire safety culture throughout all levels of plant management and operation.

This operational Safety Guide further establishes the *defence-in-depth* concept as the underlying principle for nuclear fire safety:

> This concept incorporates multiple levels of protection which are subject to various layers of overlapping provisions. These levels of protection are intended to compensate for human error or plant failures, and encompass accident prevention, mitigation and radiation protection [1].

The importance of manual fire fighting capability as a back-up method exemplifies the *defence-in-depth* emphasis on levels of protection and overlapping provisions. The Guide's requirements that personnel be qualified by education and experience to perform their duties and that all staff be trained in fire safety issues also reflect the *defence-in-depth* intentions. This philosophy underlies the requirement for a plantwide safety culture, which encourages a "rigorous approach to . . . activities and responsibilities, and . . . a questioning attitude in the performance of . . . tasks to encourage continual improvement" [1].

The *defence-in-depth* concept also is evident throughout the Guide's discussions of plant management and operational elements, particularly in its emphasis on the importance of formal procedures covering all aspects affecting nuclear safety to provide personnel with a common understanding of fire safety related issues and procedures. The Guide underscores the importance of developing a formal plant policy on fire safety and of identifying specific responsibilities and authority for all staff members involved in fire safety activities, including a formal, documented chain of command clearly communicated to all affected staff. Quality assurance and records/document management systems are required to track fire safety activities and plant modifications, and field walkdowns are recommended to verify documents and drawings.

3. FIRE SAFETY PRACTICES

The five Safety Practices developed under the IAEA Fire Safety project provide detailed instruction on implementing the requirements and recommendations of the design and the operational Safety Guides. The five Safety Practices are intended for use by trained fire safety experts to ensure an acceptable level of fire safety for systems and equipment which could potentially affect nuclear safety. Three of these documents have been published by the Agency for the purpose of conducting periodic fire safety assessments at existing nuclear power plants. The other two have been developed to assist in the development and preparation of fire hazard analyses for either new or existing nuclear power plants.

These five Safety Practices reflect the overlap which is integral to the *defence-in-depth* concept, particularly regarding data collection. The methodology of each Practice begins with data collection, which can be divided into three phases. The first phase is the review of written material and documentation supplied by the plant, including plant drawings, procedures, design basis documentation, and records. The second phase involves the discussion of specific issues with plant management, engineers, operators, maintenance staff, and fire brigade members. The third phase entails a physical inspection of all accessible plant areas to obtain a visual impression of the site and to verify information obtained in the first two phases.

In addition to the consistency which exists in the Safety Practices' common approach to data collection, some overlap exists among the types of data examined for each Practice, with varying levels of detail. Although more than one Practice may recommend the examination of a specific sprinkler system, for example, one Practice may address simple verification that the installed system is maintained and functional; and another may evaluate more detailed information, such as the system design criteria, reliability of system components, spacing of individual sprinkler heads, and the measures present to verify that there is no water flow blockage within the piping.

Each of the three Safety Practices pertaining to periodic plant assessments addresses separate aspects of fire safety. To evaluate the overall level of fire safety at a nuclear plant, all three documents must be used. These documents should be used by qualified plant personnel or by outside consultants, regulators, or safety assessors to perform inspections or audits, or they may be used by appropriate plant personnel (managers, operators) to ensure that items which may be examined during an audit are acceptable.

The key to effective employment of these Safety Practices in an assessment is to consider the assessment an opportunity for a free and open exchange of state-of-the-art fire safety issues between the inspector(s) and the plant staff. This is particularly true when the assessment is conducted by an outside consultant or fire safety expert. Open communication is essential throughout the process. It is important to remember that the objective of such an assessment is not to find fault or assign blame for identified deficiencies. Rather, the objective should always be to identify means of improving fire safety or to identify good practices which can be shared with other sites to improve their fire safety programs.

3.1. *Inspection of Fire Protection Measures and Fire Fighting Capability at Nuclear Power Plants*

Safety Practice 50-P-6, *Inspection of Fire Protection Measures and Fire Fighting Capability at Nuclear Power Plants* [2], provides detailed instruction on evaluating the design and the installation of fire protection measures within existing nuclear plants. Specifically, this Safety Practice focuses on the effectiveness of passive fire protection measures, active fire protection systems and equipment, and the manual fire fighting capability of the plant (fire brigade) to extinguish or contain fires which could adversely affect systems, equipment and components essential to safe shutdown. The passive, active and manual fire protection measures introduced in the design and operational Safety Guides are provided a more thorough discussion in this Safety Practice. For example, while the design Safety Guide requires that the "detection and alarm system shall be energized at all times" [8], the Safety Practice suggests practical ways to ensure this reliability, such as providing electrical supervision to indicate circuit faults and loss of power supply; and checking electrical interlocks associated with fixed extinguishing systems which may be used as a means of system actuation, ventilation fan shutdown, or fire damper closure.

This Safety Practice includes a detailed checklist which is intended to provide a suggested list of fire safety elements to be assessed. It is understood that some elements may be added and others deleted depending on the unique design, arrangement, construction, and operation of each specific plant. The Fire Safety Inspection Checklist is intended for use by trained experts in fire safety. The qualifications of these experts should include knowledge of fire safety design, fire protection systems and equipment, manual fire fighting methods, and nuclear safe shutdown considerations.

The inspection Safety Practice relies heavily on the requirements of the design Safety Guide. For example, when conducting an evaluation of the fire protection measures installed in a nuclear plant, it is first necessary to evaluate whether or not fire protection measures are installed in required locations and whether or not the protection provided is appropriate to the fire hazard. The design Safety Guide provides general requirements pertaining to appropriate types and locations of various types of passive and active fire protection measures. The inspection Safety Practice assesses details pertaining to specific fire safety measures, such as the temperature rating of the sprinkler heads or the obstruction of the water distribution pattern by ventilation ducts or steam piping. The documentation contained in a comprehensive fire hazard analysis of the plant and the Member State-specific regulatory requirements affecting the plant design also should be referenced to make this initial determination of appropriateness. Additional factors to consider include the overall fire safety design philosophy, plant arrangement and design features, location of critical plant equipment and essential electrical cables, and identification of plant-specific safe shutdown paths. All of this essential information should be guided by the design Safety Guide and should be documented in the plant fire hazard analysis or other similar design basis documents.

One topic addressed by the inspection Safety Practice that is not covered in the design Safety Guide is the use of qualified equipment and components. A design requirement common to all fire protection measures and systems is the need for a method of equipment qualification. For example, specific fire safety components should be tested and approved specifically for fire service use by an independent testing laboratory, or third party certification should be provided to document equipment reliability. This issue applies to passive fire protection measures such as fire doors and fire rated dampers, to fire detection and alarm systems, and to extinguishing system components and equipment.

3.2. *Assessment of the Overall Fire Safety Arrangements at Nuclear Power Plants*

Safety Practice 50-P-11, *Assessment of the Overall Fire Safety Arrangements at Nuclear Power Plants* [3], provides instruction and a detailed checklist for assessing all the fire safety arrangements of a nuclear plant except for the design and installation of passive and active fire protection measures, manual fire fighting, and the fire hazard analysis. The document defines *overall fire safety arrangements* to mean "an integrated system of organization, management, procedural controls, [routine maintenance of] passive and active fire protection measures, quality assurance and records management systems related to plant fire safety" [3]. Given this scope, this Safety Practice essentially is designed to assess the plant's safety culture. Safety culture, as described in the operational Safety Guide, promotes "a continuing awareness of the importance of preventing fires in nuclear power plants" and is influenced mainly by "the organizational framework set down by the plant management, and the attitude of staff operating within that framework" [1].

This Safety Practice provides a means of assessing the organizational framework developed by plant management to foster an awareness of the importance of fire safety within the plant and the specific measures established to ensure effective implementation of fire safety. It is written for operators, regulators and safety assessors; and it extends its scope beyond concern solely with safe shutdown to include "the protection of site personnel, the public and the environment from undue radiation hazards" [3]. The document contains details to aid in the assessment of programmatic aspects of nuclear power plant fire safety pertaining to fire prevention; administrative and procedural controls; and the periodic inspection, maintenance and testing of installed fire protection features. This organizational Safety Practice echoes the operational Safety Guide's emphasis on formal fire safety policies and procedures, clearly defined responsibilities and authority for staff involved in fire safety activities, quality assurance and records/document management.

The detailed Fire Safety Inspection Checklist included in this Safety Practice addresses topics discussed in the operational Safety Guide to ensure that policies and procedures are in place which address fire safety organization and management; engineering review of design changes and modifications; control of combustible materials; control of ignition sources; inspection, maintenance and testing of the fire protection measures (both passive and active); records and documentation; and quality assurance. The first part of the checklist (sections A through G) applies to a procedural review of these areas; and the second part (section H) applies to a field walkdown to verify that actual conditions are consistent with the written documents and to gain an impression of overall effectiveness of the program implementation.

This Fire Safety Inspection Checklist overlaps fire safety elements discussed in the other two assessment Safety Practices. In addition to verifying the adequacy of the routine inspection, maintenance and testing programs for fire detection and extinguishment systems, the checklist includes direct observation of these systems as installed in the plant. The difference between this checklist and the one contained in the inspection Safety Practice is that this checklist assesses the ongoing maintenance, operability and effectiveness of these systems, and the other addresses issues such as the specific standards to which systems were initially designed and installed. An efficient method of using the organizational Safety Practice checklist would be to examine the inspection Safety Practice checklist along with the other documents reviewed in the initial phase of the assessment, before conducting the walkdown.

During the field walkdown portion of the assessment of the overall fire safety arrangements of the plant, particular emphasis should be placed on those areas containing equipment important to plant nuclear safety. This equipment should be documented in the fire hazard analysis. During the physical walkdown of the plant, the inspector also should determine compliance with information contained in the fire hazard analysis, particularly with respect to fire loads in specific plant areas. Again, the *defence-in-depth* concept is evidenced by some overlap between fire safety elements assessed using the organizational Safety Practice and those addressed in the fire hazard analysis.

3.3. *Evaluation of Fire Hazard Analyses for Nuclear Power Plants*

Another aspect of nuclear plant fire safety that should be evaluated is the existing fire hazard analysis. Safety Series 50-P-9, *Evaluation of Fire Hazard Analyses for Nuclear Power Plants* [4], provides a methodology to be used for such an evaluation. Preparation of a new fire hazard analysis is covered by another Safety Practice (discussed in Section 3.4 of this paper). This Safety Practice can be used by regulators or independent safety assessors to evaluate the adequacy of the existing fire hazard analysis, or it may be used by plant personnel for guidance in performing an in-house assessment of the plant's fire hazard analysis. Guidelines described in this publication should be used by trained fire safety experts who have a working knowledge of fire safety design in nuclear plants; thorough familiarity with fire protection systems and equipment available for use; knowledgeability in nuclear plant operation, including safe shutdown considerations; and proficiency in the methods used to quantify fire hazards.

This Safety Practice is designed to assess the adequacy of the fire hazard analysis in terms of the objectives outlined for such an analysis in the design Safety Guide. Essentially, these objectives entail the assessment of how well the fire hazard analysis addresses the "identification of fire hazards and safety systems; analysis of fire growth; and adequacy of fire protection" [4]. The scope of the fire hazard analysis is to assess all areas of the plant to ensure that adequate fire protection is provided for systems and equipment essential for safe shutdown and for those which could, in the event of a fire, cause a radiation hazard to site personnel, the public or the environment. The fire hazard analysis is basically comprised of three parts: identifying all fire hazards and safety systems, analyzing fire growth, and assessing the adequacy of the fire protection measures. Common elements within all fire hazard analyses should include at least the following steps:

(a) Identify the location of and describe important safety systems.
(b) Document the specific combustible fire load in each fire compartment.
(c) Document all installed fire protection measures (both active and passive measures).
(d) Identify the design basis fire scenario for each fire compartment.
(e) Identify the approach used to quantify fire growth in each fire compartment, including the consequences for nuclear safety systems.
(f) Verify the adequacy of fire protection measures to ensure plant nuclear safety.
(g) Update the fire hazard analysis periodically to reflect plant changes.

The Safety Practice recommends field walkdowns to check the accuracy of the fire hazard analysis in reflecting actual conditions. It also recommends that the fire hazard analysis take into consideration the differences in design, construction, and fire safety philosophy among nuclear power plants. All of these factors play a part in determining the level of fire safety needed for the specific plant being analyzed. The fire hazard analysis evaluation must include a determination of whether the analysis is using the fire containment approach or the fire influence approach. These different approaches are defined in the Agency's design Safety Guide as follows:

> As a result of the integrity of the fire barrier around each fire compartment, the spread of a fire from one compartment to another is prevented and thus the concept of segregation of items important to safety can be achieved for the periods specified for the fire barriers. This configuration is called the fire containment approach. However, in certain fire compartments, the spread of fire within the compartment may also have to be prevented in order to limit the impact of fires on items important to safety. In such cases active fire detection, extinguishing or passive means, in conjunction with the provision of appropriate distances between components is used to prevent the spread of fire from one fire cell to another within the fire compartment. Such a configuration is called the fire influence approach [8].

The approach used is dependent largely on plant design and will determine the types of information that should be included in the fire hazard analysis. When the fire containment approach has been used, for example, it is necessary to determine that the analysis has appropriately addressed the need for adequate separation of redundant safety systems and the possibility of fire exposures from adjacent fire compartments. If the analysis has used the fire influence approach, the evaluation of the analysis must determine if it has considered other issues, such as the effects of heat and smoke within a fire compartment, the potential for flammable liquid or gas spread within a fire compartment, and the spurious activation of fire extinguishing systems.

Analyzing fire hazards at nuclear power plants is an iterative process. The ultimate objective of evaluating the fire hazard analysis is to ensure that fire hazards have not been overlooked and to verify that fire protection measures are appropriate to ensure plant nuclear safety. If the evaluation of the fire hazard analysis identifies problem areas, recommendations should be generated and the fire hazard analysis process repeated to address the identified problems. A successful fire hazard analysis will ensure adequate fire protection levels for nuclear safety systems, develop recommendations for any problem areas, repeat the fire hazard analysis process to address these areas, and verify that the fire hazard analysis is updated periodically to reflect design changes and modifications.

3.4. *Preparation of Fire Hazard Analyses for Nuclear Power Plants*

The draft Safety Practice entitled, *Preparation of Fire Hazard Analyses for Nuclear Power Plants* [5], provides a framework for the preparation of a fire hazard analysis for a new or an existing nuclear power plant. This publication should be useful to plant designers or operators for the preparation or updating of a fire hazard analysis; but it is intended primarily for use by trained fire safety experts who have knowledge and experience in fire safety systems and equipment, fire growth quantification and analysis, computational methods for predicting fire consequences, and nuclear power plant operations and safety systems. Due to the wide variety of knowledge and experience needed to perform this comprehensive analysis and the length of time required to accomplish the task, it usually is recommended that a team approach be used in preparing a new fire hazard analysis.

This Safety Practice includes much more detail about the contents and methodology of a fire hazard analysis than the Practice which covers evaluating such an analysis. For example, this Safety Practice provides instruction on applying the quality assurance program to the fire hazard analysis, subdividing buildings into fire compartments and fire cells, determining whether to use the fire containment or fire influence approach, collecting data, analyzing fire growth and potential consequences, and determining when to update the fire hazard analysis. The discussions in both documents concerning the use of appropriate personnel and the scope and purpose of a fire hazard analysis are very similar.

For existing nuclear power plants, the purpose of the fire hazard analysis is to document that existing fire protection measures are adequate to ensure plant nuclear safety (as defined in item 6 below). For the design of new plants, the fire hazard analysis has several purposes, which this Safety Practice excerpts from the design Safety Guide:

1) *To identify items important to safety and their location.*
2) *To analyse anticipated fire growth and the consequences of the fire with respect to items important to safety.*
3) *To determine the required fire resistance of fire barriers.*
4) *To determine the type of fire detection and protection means to be provided.*
5) *To identify cases where additional fire separation or fire protection is required, especially for common mode failures, in order to ensure that items important to safety will remain functional during and following a credible fire.*
6) *To verify that the intent of paragraph 216 of the Safety Guide 50-SG-D2 (Rev.1) has been met. Paragraph 216 states "The safety systems required to shut the reactor down, remove residual heat and contain radioactive material shall be protected against the consequences of fires so that the safety systems are still capable of performing the above safety functions, taking into account the effects of a single failure as required in the Code on Design for these functions".* [8]

For either existing or new plants, in situations where deficiencies are identified during the analysis, the process requires recommendations to be formulated which, when implemented, will ensure that plant nuclear safety is achieved.

This type of fire hazard analysis (also known as a deterministic fire hazard analysis) is a systematic process which should include the following six steps:

(a) data collection
(b) analysis of fire growth
(c) consequence analysis
(d) evaluation of adequacy of fire protection (as defined by the purpose stated above)
(e) recommendations for improvement
(f) repetition of analysis, as needed

The data collection step is a lengthy and time-consuming process, but it is essential to the accuracy of the final analysis. This Safety Practice provides detailed instruction on conducting the inventory of all nuclear safety systems in the plant, including safety equipment components and electrical cabling (control, instrumentation, and power cabling) and interactions between safety systems and cabling to various safety system components. The document also provides details to assist in the inventory of the combustible fire load, potential sources of ignition, passive fire protection measures, equipment available for manual fire fighting, and the extent to which the overall fire safety design philosophy for the plant relies on manual fire fighting as a primary means of fire protection.

The section of this Safety Practice which describes the analysis of fire growth includes guidance on determining the mechanism of fire ignition and fire growth rate and identifying potential fire scenarios for each fire compartment. The section which follows offers guidance on performing a consequence analysis to determine the impact of the fire on the safety systems in the area. The objective of this analysis is to determine, for each fire compartment, whether or not the fire will threaten the ability to achieve safe shutdown, including residual heat removal, by simultaneously disabling redundant equipment which is part of a safety system. Figure 1, which

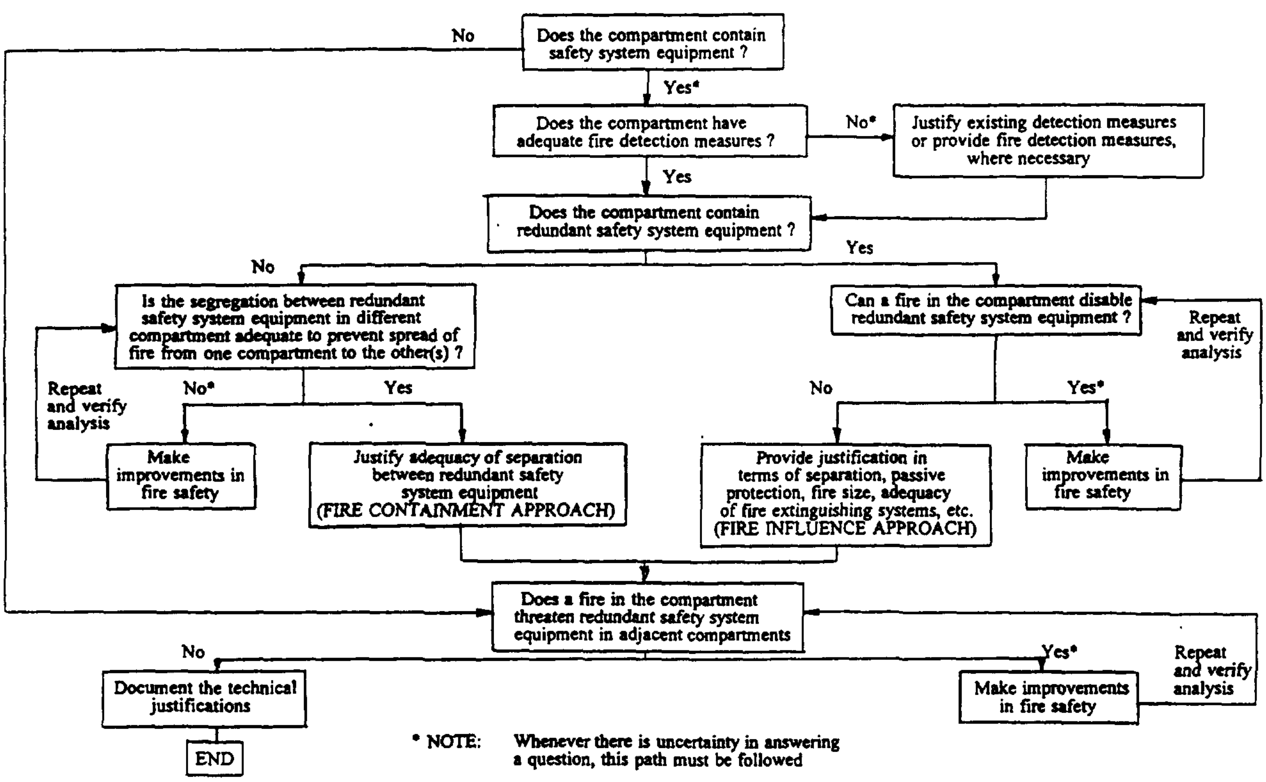

FIG. 1. Assessment of fire consequences for a fire compartment.

outlines the principal stages of the consequence analysis process, is included in this Safety Practice.

In addition to such step-by-step instruction, the consequence analysis section of the Practice provides guidance on the three distinct methods which may be used to determine the effect of fire on safety systems:

(a) Subjective evaluation—which relies on the application of practical experience and engineering judgment to assess the adequacy of the fire safety systems and which generally is used only as a screening tool or to gain a conservative assessment

(b) Hand calculations—which may be used where a relatively simple fire compartment is being evaluated to quantify fire growth rate, fire duration, smoke layer depth, radiant heating, etc., by using formulae, graphs, tables, or empirical correlations published in the generally accepted literature

(c) Computer fire modeling—which may be useful for determining peak temperatures, maximum rates of temperature rise, and actuation times for fire detection and extinguishing systems

These methods may be used either alone or in combination, depending on the circumstances and the complexity of the fire compartment. The analysis should include detailed evaluation of direct, indirect, and secondary effects of fire, which often are not obvious. Specific examples of secondary and indirect fire effects are provided in the appendix to the publication.

The judgment of a fire safety system's adequacy should be technically justified and documented and should allow sufficient detail such that independent assessors of the work can understand how the judgment was reached and can recognize that a systematic approach was used throughout the analysis. The Safety Practice contains advice about recommendations for improving fire safety systems, including considerations of other plant systems capable of achieving the desired safety functions, as well as the installation of a new, independent safety system to accomplish the needed function. This Practice reiterates the recommendation found in the Safety Practice for evaluating fire hazard analyses which concerns the need for repeating the fire hazard analysis to ensure that fire safety objectives can be met after inclusion of the new enhancements.

3.5. *Treatment of Internal Fires in Probabilistic Safety Assessments for Nuclear Power Plants*

The draft document entitled, *Treatment of Internal Fires in Probabilistic Safety Assessments for Nuclear Power Plants* [6], is a guide for preparing a fire probabilistic safety assessment (PSA) for a nuclear plant. Its intended users are "professional staff managing or performing PSAs" [6]; and it advises that a fire PSA will require a multidisciplinary team of plant personnel knowledgeable about the plant design and operation, PSA techniques, fire science, and fire protection systems as well as the interaction of such systems with nuclear safety systems. The document refers to two other NUSS publications which address probabilistic safety assessments for nuclear power plants: one for external events and the other for internal events. This Safety Practice is intended to supplement the latter publication.

The fire PSA is similar to the deterministic fire hazard analysis in that both employ a systematic assessment of all areas of the plant which entails data collection and the definition of fire compartments and cells. Usually, the fire PSA builds upon and supplements information found in the deterministic fire hazard analysis. As such, it provides a greater level of detail and applies different acceptance criteria for the assessment of fire safety. Specifically, the fire PSA introduces the concept of the *likelihood* of fire to occur in a given location in the plant and the *probability* that the fire will grow to an extent that it can damage plant safety systems. This Safety Practice differs from the others discussed in this paper in that its emphasis is on "assessing the potential risk of core damage states initiated by fires" [6] using a probabilistic approach. The Safety Practice recommends screening out fire scenarios which do not pose a probable risk of core damage.

This screening process begins after data collection and fire compartment definition with an examination of the existing internal events PSA. The fire PSA relies on the plant response model developed for internal initiating events; therefore, one objective of this examination is to determine which of the logic models (e.g., event trees, fault trees) used for the internal events PSA are applicable to the fire PSA. Another objective is to identify PSA related equipment and cables in each fire compartment, information which must be verified by a plant walkdown. Screening of fire scenarios is performed according to potential impact and probable frequency. For the remaining fire scenarios which have not been screened out by impact or by frequency, a quantitative PSA model is used for further analysis.

The Safety Practice provides detailed discussion about screening techniques, as well as the type of information to consider during data collection, fire compartment definition and internal events PSA examination. It also provides guidance on the actual detailed analysis, which often involves quantitative fire modeling and which is performed with the objective of reducing excessive conservatism which may remain for fire scenarios not screened out. This section of the Safety Practice provides guidance on integrating other factors into the analysis, such as the effect of thermal fire barriers and other fire protection measures and the specific location of equipment within fire compartments. The document contains references to realistic models which can be applied for the assessment of human actions, fire propagation, and the effects of fire on equipment and cables to determine equipment damage thresholds. This detailed analysis is intended as an extension of the screening process in order to determine the most realistic (i.e., probable) fire scenarios and plant responses.

To aid in this determination of the most realistic scenarios, the Safety Practice includes brief discussions of issues which have proven specifically pertinent to PSAs. These include the control room, cable spreading room and other sensitive plant locations, environmental survival of equipment, fire induced explosions, control systems interaction, containment integrity and guidance on performing PSAs with incomplete information. This discussion of the incompleteness of information used for the assessment is expanded in the following section, which addresses the degree of uncertainty inherent in the fire PSA results. For those fire scenarios which remain after the screening process and detailed analysis, the Safety Practice recommends additional analysis to evaluate sources of uncertainty and to determine how sensitive the fire PSA results are to the input data, models and assumptions. The PSA model introduces elements involving statistical data; therefore, additional contributors to uncertainty are inherent in the final evaluation of fire safety. Because of this aspect, uncertainty and sensitivity analysis is necessary for the correct interpretation of results.

Like the Safety Practice pertaining to preparing a deterministic fire hazard analysis, this Safety Practice contains guidance on documenting and applying quality assurance to all the phases of the analysis and on determining the format, content, and level of detail of the final report. This Safety Practice also includes a number of appendices which provide clear examples of essential equipment malfunctions which can be caused due to fire induced damage in nonessential circuits such as low impedance faults, high impedance faults, and secondary ignitions in non-essential cables. Additional appendices provide examples of physical propagation of fire between nonessential and essential equipment and secondary ignition of fire due to short-circuit overcurrent in nonessential equipment. Appendices describing examples of detailed analytical methods and fire propagation event trees provide further guidance on performing these elements of a fire PSA.

4. TECHNICAL DOCUMENT—*Guidelines for IAEA Fire Safety Review Services*

The final document to date in the Agency's Fire Safety project, entitled, *Guidelines for IAEA Fire Safety Review Services* [7], provides guidelines for the organization and conduct of fire safety review missions under the IAEA Engineering Safety Review Services program. This TECDOC contains guidance for defining the overall scope of an IAEA-sponsored fire safety review mission, be it a comprehensive review of all elements of fire safety at a nuclear power plant, a focused review of a specific topic such as manual fire fighting capability, or a combination of elements chosen to best suit the specific needs of the Member State. This document is intended primarily for IAEA staff and other experts involved in IAEA-sponsored fire safety review missions and for Member States interested in such a review or in conducting internal plant reviews.

IAEA-sponsored fire safety reviews are conducted according to the requirements and recommendations contained within the Safety Guides and Safety Practices discussed in this paper. The section of this TECDOC which indicates the IAEA documents appropriate to each type of review provides a concise summary of the fire safety elements which are addressed in each of the Safety Practices. In addition, this Technical Document provides a list of the documentation needed from each plant for each type of review. Because English is the working language of all fire safety review missions, some guidance is provided on the extent of translation which may be required of some plant documents.

This document provides detailed guidance on how to organize and plan a fire safety review mission, including scope, level of detail, schedule and number of fire safety experts needed to perform the work. It outlines what should be accomplished at the preparatory meeting, such as a clearly defined scope of work; and it addresses contextual issues such as tailoring the scope to fit the needs of the Member State and understanding the regulatory regime of the Member State. This publication also discusses the responsibilities of the IAEA and Member State representatives during the conduct of the mission. Daily meetings between IAEA reviewers and plant personnel are encouraged, in addition to entrance and exit meetings.

The fire safety review methodology recalls the basic data collection methodology outlined in the Safety Guides and Practices discussed in this paper: document review, personnel interview, and direct observation of site conditions. This TECDOC provides a discussion of what information should be contained in the final report. A suggested report format is also provided in the appendix.

5. SUMMARY AND CONCLUSION

The series of IAEA fire safety documents discussed in this paper is intended to provide guidance to Member States for the enhancement of fire safety within their nuclear power plants. The draft operational Safety Guide expands upon the design Safety Guide by providing further guidance for managers and operators of existing nuclear power plants in achieving improved levels of fire safety during the day-to-day operations at the plant. The Safety Practices provide a detailed framework for evaluation—and implementation—of all fire safety measures in an existing nuclear power plant, including deterministic and probabilistic assessment tools (i.e., fire hazard analysis and fire PSA). The Technical Document discusses IAEA-sponsored fire safety reviews which are conducted in accordance with the Safety Guides and Practices. The levels of overlap evident among these documents are intentional, intended to reinforce the *defence-in-depth* approach to nuclear safety. The common objective of all of them is the prevention, detection and extinguishment, and containment of fires which may affect safe shutdown or effect radioactive releases.

Even though several documents in the IAEA fire safety series are still in the final stages of review or publication, the material is available and has been used by a number of Member States and by existing nuclear power plants in many different countries. The results of a recent informal survey indicate that all IAEA fire safety documents have been used to some degree (even those not yet officially published). Some groups are using the publications primarily as a reference guide while a few others have used them in their periodic assessment activities. Almost half the respondents to the survey have used the Safety Practice 50-P-9 entitled, *Evaluation of Fire Hazard Analyses for Nuclear Power Plants.* This is the most frequently used document in the series at the present time. Another third of those responding have translated at least a portion of the documents into their native languages. Virtually all respondents indicated that they planned to use the Agency publications for future fire safety assessment activities.

Such wide use of these IAEA fire safety documents, even in draft form, seems to indicate a recognition of the importance of fire safety within nuclear power plants, as well as the importance of appropriate guidance on the implementation of fire safety measures within these plants. As the availability of these publications becomes more widely known, it is hoped that the information contained in these documents will be used in the industry to effect significant improvement in the overall level of fire safety at existing nuclear power plants throughout the world. At the very least, it is hoped that the levels of overlap and the vigilance urged within these documents serve as a constant reminder of the phrase found in the design Safety Guide [8] which summarizes the Agency's overall fire safety philosophy—*Safety is not a static concept.*

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Invited Paper

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LESSONS LEARNED FROM IAEA FIRE SAFETY MISSIONS

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Abstract

The IAEA has conducted expert missions to evaluate fire safety at the following nuclear power plants: the Zaporozhe plant in the Ukraine, the Borselle plant in the Netherlands, the Medzamor plant in Armenia, the Karachi plant in Pakistan, the Temelin plant in the Czech Republic, and the Laguna Verde plant in Mexico. The scope of these missions varied in subject and depth. The teams sent from the IAEA consisted of external fire experts and IAEA staff. All the missions were of great use to the host countries. The participating experts also benefited significantly. A summary of the missions and their findings is given.

1. THE IAEA FIRE SAFETY MISSIONS

1.1. Nature of the Missions

Several years ago, the IAEA initiated the Engineering Safety Advisory programme to provide advise and assistance to Member States to enhance safety in a number of selected disciplines. Fire protection is one of these disciplines. Under this programme, the Agency has conducted fire safety missions to nuclear power plants, upon the request of some Member States. To date, six missions had been carried out. These plants were of different design and in different stages of construction or operation. The scope of the missions were determined by the needs of the nuclear power plants and were established in advance. Each mission was carried out by a team (hereafter in this paper called the Team) consisting of external experts on assignment to the IAEA and usually includes an IAEA staff member. The external experts were recruited from various countries based on their experience in fire protection in nuclear facilities. They came from Finland, France, Spain, Canada, Germany, India and USA.

1.2. Evaluation Methodology

The evaluation was based on the IAEA Safety Series No. 50-SG-D2 (Rev. 1) "Fire Protection for Nuclear Power Plants". In addition, several Safety Practice documents on fire protection produced by the Agency were also used for guidance. Learning from earlier missions, some areas of difficulties were gradually eased in later missions, for example:

1) There was insufficient time to gather the information needed, including review of documents, interviews with counterparts and doing plant inspections. Later missions were changed from one week to two week in duration and the scope of review were more clearly defined. Whenever practical, documents were sent to the experts prior to the missions.
2) The speed of the evaluation was often slowed by language barriers, particularly when reviewing documents in a language foreign to the Team members. Whenever practical, translation of key documents into English was requested. In all cases, interpreters were available throughout the mission.

3) The IAEA Safety Series was written for new plant design. Plants of older design often employed very different principles and means in fire protection. While the installed features of a plant were compared to requirements in the Safety Series, the intention was not to provide a pass-fail type of result. The Teams made judgements on the significance of the differences and the practicality of back-fitting. Alternative protective measures were also explored.

The Fire Safety Missions provided an opportunity for the Teams to explore, with the counterparts, realistic steps by which fire safety could be improved at each plant. As the evaluations were dependent heavily on judgement, results of different missions should not be compared.

Learning from the experience of the completed missions, an IAEA TECDOC "Guidelines for IAEA Fire Safety Review Services at Nuclear Power Plants" was produced to provide more formal guidance for future missions. This document is described in another paper at this Symposium.

Results of the missions were published in IAEA documents. These are listed in the Bibliography. Summaries of the missions are presented in the following Sections.

2. ZAPOROZHE NPP, UKRAINE, 1993 AUGUST 9-13

The Zaporozhe Plant consists of six units with the WWER-1000 reactors. At the time, five units were in operation with the sixth ready for fuel loading. The duration of this mission was one week. The scope included a training workshop and a broad evaluation of the plant documentation and fire protection features.

The Team of three external experts and two IAEA staff members delivered the training workshop which covered fire safety principles, basic requirements, fire protection programmes, fire hazard analysis, active and passive fire protection measures. An inspection of selected plant areas was conducted. Discussions were held with the plant chief engineer, deputy chief engineer for safety, fire protection engineers and other counterparts. The Team was given a presentation on the national fire regulations, the Zaporozhe plant fire protection concepts and fire protection features, plant organisation, administrative controls and maintenance procedures. The major insights and conclusions from the mission were:

1) The Deputy plant manager was responsible for safety. There were 2 fire protection engineers and an advisory Fire Technical Committee in the plant. There was an on-site fire brigade that reports independently to an external organisation. The Team found that the organisational structure responsible for fire safety was not clear and not well documented. Preparation of a comprehensive fire protection programme was recommended.
2) Past fire incidents in the plant included a major fire on turbine building cable trays in Unit 1 prior to start of operation and a transformer fire in Unit 2 causing some damage to the electrical building.
3) The plant fire protection design was done to different revisions of former USSR Standards. A review of the design against currently applicable standards was recommended.
4) All cable duct areas, spreading rooms and electrical power distribution rooms have fire detectors, mostly of the combined smoke and heat type, connected to an alarm panel near the

control room. The availability of the fire detectors was not monitored and a recommendation was made to do so.

5) There was a site fire water system and a safety-related fire water system with three 100% pumps and tanks. The pumps were supplied with emergency power backup.

6) The three trains of reactor safety systems were well separated physically except at the cable spreading rooms where several improvements were recommended. The Team also noted that throughout the plant fire doors were not labelled and many were poorly maintained and left open.

7) There were no fire detectors or fixed fire suppression systems for the reactor coolant pumps. Fire detectors and oil leakage collectors were recommended.

8) The plant had only a brief fire hazard analysis was performed based on a plant walkdown and broad judgement. The experts and counterparts all agreed that a more detailed fire hazard analysis was necessary.

9) Additional fire detectors were recommended for the diesel generator building, the safety pumps area and the transformers.

10) A full-time, in-plant fire brigade was available to assist plant operators in case of fire. Drills showed that arrival time at the plant was within 5 minutes. Training and equipment appeared adequate.

11) Plant management was serious about improving fire safety as evidenced by the modifications already executed or planned.

3. BORSELLE NPP, NETHERLANDS, 1994 OCTOBER 24-28

The Borselle NPP is a single unit PWR of Siemems design with a power output of 480 MWe. It has been in commercial operation since 1973. The main objective of the mission was to review the Fire Hazard Analysis (FHA) that was just completed. The review also extended into the plant fire protection programme and the installed fire protection features. The experts were sent translated excerpts of the FHA in advance. During the one-week mission, they held discussions with the counterparts, reviewed documents and performed a brief plant inspection. The major insights and conclusions of the mission were:

1) Overall responsibility for fire safety rested with the site vice-president. Functional responsibilities related to fire prevention, fire protection system maintenance and fire fighting were split between two operations departments. The in-plant fire brigade was composed of trained operating personnel. The Team felt that a better co-ordination of efforts could result if these responsibilities were more clearly defined and documented.

2) Records for the past ten years showed 3 small fire incidents and 15 events involving degradation /failures of fire systems.

3) Review of the FHA was limited to the general methodology plus some sample fire compartments. The FHA did not go into quantification of fire growth and compartment conditions. The qualitative evaluation was based on engineering judgement. The Team asked for some examples and found the arguments to be generally sound. However, proper documentation of the assumptions and steps in the judgement process should have been documented. The Team further commented that transient combustibles should have been considered and the Emergency Lighting System and Emergency Communication System should have been included in the FHA.

4) Housekeeping at the plant was acceptable but minor improvements were recommended. Structures and equipment were in good condition considering the age of the plant.

5) The fire alarm system was a modern, solid state system with addressable detectors which replaced the original equipment. Fire suppression systems included sprinklers, water spray systems and a large number of halon and carbon dioxide systems. Halon was not being replaced as it continued to be available from a banking system. The Team recommended installing fixed suppression systems in the cable cellars in the switchgear building and simplifying the manual control of the water spay systems. Two fire water systems were available: a low pressure system for the whole plant, with one diesel pump and one electric pump; a high pressure system for protecting the reactor coolant pumps, with two redundant electric pumps. The fire systems were tested and maintained properly.

6) All boundary walls of a fire compartment in the plant had 60 minutes of fire resistance. Many doors of the old type had only 30 minutes fire resistance. Adequacy of these barriers should be determined in the FHA. The Team believed that at least the doors between the turbine building and switchgear building should be upgraded to a higher fire resistance. Because of the age of the design, redundant trains of safety related systems were not always in separate fire compartments. The plant was considering modifications to improve physical separation between the two trains of the emergency power systems and the high pressure injection systems. These modifications were supported by the Team.

7) The in-plant fire brigade consisted of a minimum of six trained operating persons at all times plus 3 full-time firemen on duty during the week days.

Outside professional fire departments from nearby towns will respond when called. In general, the organisation, training and equipment of fire response forces appeared acceptable. Some suggestions on improving the pre-fire plans were made.

4. MEDZAMOR NPP, ARMENIA, 1994 NOVEMBER 11-16

The Medzamor NPP has two units each with a WWER 440/230 type reactors. Commercial operation began in 1976 and 1980. The plant had been shutdown after a severe earthquake in 1988. The mission was carried out by three external experts plus an IAEA staff member. The mission involved a combination of document review, discussions with counterparts and field inspection.

A fire in 1982 inside a cable tunnel which disabled power supply and I & C systems of both units led to significant modifications to improve fire protection. These included putting fire resistant coating on the PVC cables, extending the foam system to cover more rooms, installing hydrants on the turbine building roof, upgrading of fire detectors, installing 45 minute fire doors, improved joints on oil piping, improved access for fire fighting in the turbine building, and installing independent power cables to the room housing safe shutdown equipment. Further upgrading was planned, the included: installing fire detectors in all rooms containing electrical equipment, installing an independent, remote shutdown panel, replacing the foam systems with sprinklers and dividing the cable tunnels with fire walls. The major insights and conclusions from the mission were:

1) Overall fire safety was supervised by the Chief Engineer of the plant. Responsibilities were shared by different operational groups and the on-site fire brigade. There was no special organisational structure responsible for overall fire safety. The Team recommended increasing the authority of the Senior Engineer on Fire Safety and establishing a more comprehensive program to carryout and co-ordinate all shared fire protection responsibilities.

2) During the plant inspection, the Team noticed some transient combustibles and oil spills, and extensive uses of loosely laid PVC floor covering. Elimination of these combustibles was recommended.

3) The fire detection system consisted of a mix of old and new detectors. They did not cover all areas with safety important systems. Since the plant placed a high reliance on manual fire fighting, the fire alarm system had an important role. Recommendation was made for installation of a new fire alarm system.

4) The plant had extensive use of foam systems. Replacement of these systems by sprinkler systems was recommended for the cable rooms. Manual fire fighting was relied on in most plant areas but equipment did not appear to be adequate.

5) Redundant safety equipment and cables were not always provided with adequate physical separation. Equipment with fire hazard (e.g. reactor coolant pumps) were not enclosed in fire compartments. Fire doors were not qualified with fire resistance and structural steel was not protected by barriers. Reliance was placed on fire suppression in these locations. The Team recommended upgrading the barriers to the turbine building, between the diesel generators, to the control room, and the cable rooms and tunnels.

6) Fire fighting was provided by an in-plant brigade of trained staff and an on-site full-time fire brigade. The full-time brigade also performed duties of fire safety inspection. Further support and equipment was available from off-site fire departments. The Team recommended better procedures, personal protection and communication equipment for all the in-plant brigades.

5. KARACHI NPP, PAKISTAN, 1995 JUNE 11-17

The Karachi NPP (KANUPP) consists of a single unit CANDU reactor with a gross power output of 125 MWe. It has been in commercial operation since 1972. The mission was carried out over a one week period by two external experts. The main focus of the mission was to evaluate the status of the fire detection and alarm system and specifications for a new system. The Team also held extensive discussions with the counterparts over various aspects of fire safety, reviewed some of the operational documents and did a plant inspection. The major insights and conclusions from the mission were:

1) Overall fire safety was supervised by the manager of Quality Assurance Division who reported directly to the station manager. The various responsibilities were shared between the Industrial Safety Group and the Maintenance Division. There were no specific regulatory requirements for fire protection in the nuclear power plant. A comprehensive fire protection programme was recommended.

2) Since plant commissioning, there had been 12 fire incidents of which 2 were of some significance.

3) Improvements in fire protection since commercial operation included replacement of the fire ring main, addition of hydrants and hose stations and addition of a fire alarm system in the warehouse.

4) A detailed fire hazard analysis had not been done on the plant. A room-by room review was done when specifications for the new alarm system were prepared and this could be used as the basis for a more detailed analysis. The Team discussed with the counterparts essential elements of a fire hazard analysis.

5) Housekeeping was acceptable and the plant was generally in good repair. Some transient combustibles were found. Some combustible cable penetration seals were noted and a recommendation was made to identify and replace all such seals.

6) Plant management had recognised the need to replace the old fire alarm system with a modern one. The Team reviewed in-depth the specifications for the new system and found the approach to be correct. Specific comments were made in some areas, such as use of addressable detectors, detector selection and location.

7) The lack of means to suppress fires inside the reactor building was a concern to the Team. As a minimum, large capacity dry chemical extinguishers were recommended. Additional measures such as oil collectors on the reactor coolant pumps and fire barriers to protect essential cables may be required as determined in the fire hazard analysis.

8) Brief observation showed there was the potential for a fire in the turbine building to cause loss of normal and backup power buses, resulting in power interruption to a large number of plant systems. Such an event should be assessed in detail in the FHA. Improvements including use of fire barriers to protect redundant power buses, were recommended.

9) Water spray systems were recommended to protect the cables above the distribution room and in the tunnel to the reactor building.

10) Training of the in-plant fire brigade did not appear to be adequate to address all potential fires. Improvements were discussed, including using full-time fire fighters, training in fire fighting strategies, drawing up pre-fire plans and doing drills in various plant areas.

11) The fire water supply system was from sea water with a backup at a smaller flow rate from the demineralized water tank. The system was placed on manual operation due to concern for corrosion. The Team concluded that the system was inadequate and several improvements were recommended.

6. TEMELIN NPP, CZECH, 1996 FEBRUARY 4-14

The Temelin NPP has two units with WWER pressurised water reactors each having a power output of 1500 MWe. At the time both units were still under construction. The mission spanned two weeks and was carried out by two external experts and one IAEA staff member. It included review and discussions in the office of the State Office of Nuclear Safety (the Regulatory body) in Prague and a two-day visit to the site. The scope of the mission covered the regulatory practice as well as plant design and preparedness for operation. The major insights and conclusions from the mission were:

1) General regulatory criteria on fire protection in nuclear power plants in Czech was laid down by the State Office of Nuclear Safety (SONS). The older Soviet design was abandoned in favour of adopting the IAEA Safety Series 50-SG-D2. Many of the standards used for building fire safety and fire protection systems at Temelin were taken from Czech civil codes. In some cases, manufacturers' specs and international standards were used. Both the SONS and the fire department in the District were involved in reviewing fire protection design and ensuring operational safety. The regulatory requirements and process seemed to be well in place. Recommendations offered to the SONS include: formally documenting the applicability and exceptions of codes and standards, set a position on the role of fire PRA, and to develop technical specifications on fire protection systems.

2) Forty one fires occurred during the operation of the two plants in the former Czechoslovakia Republic. Of these, 10 events had impact on safety related items.

3) In terms of the design, construction and commissioning process, the Team recommended the establishment of a project co-ordinator on fire protection to oversee all inter-disciplinary aspects.

4) The plant had a full-time, on-site brigade of 60 persons, arranged into 20 per shift. They were stationed 500m from the generating units. Equipment for the brigade, including aerial trucks, pumpers and foam tanks were adequate for the fire hazards. This full-time brigade

functionally reported to the plant. Members of the brigade also took on fire prevention duties inside the plant. The protocol was well established for involving public fire departments. Plant operating personnel were also trained and expected to respond to fires. The Team recommended a more clearly defined role and procedures for this group.

5) The fire hazard analysis was reviewed. In the analysis of fire growth, the fuel loads were compared with the load that would contribute to a standard 90 minute fire exposure to show adequacy of the barriers. Consequences of failures of the barriers (e.g. open door or failed penetration seals), effects of smoke and consequential failures were not analysed. The Team felt that in some areas the above should have been considered. Reliance was placed on the design of fire compartments to ensure sufficient systems would remain available even if everything was consumed within each fire compartment, however, the FHA did not identify the systems that would be damaged and the adequacy of the remaining systems on a zone by zone basis.

6) The Team briefly reviewed the fire PRA which was still in draft form. The preliminary results showed that fire was the largest contributor to core melt of all external events.

7) Fire detection was provided by an extensive modern system with addressable devices. Fire suppression systems consisted of sprinkler, deluge, carbon dioxide and foam systems. On site water supply for fire protection was considered adequate.

8) In general the Team was satisfied that the regulators, designers and operators had sufficient fire protection expertise and a keen desire to address fire safety.

7. LAGUNA VERDE NPP, MEXICO, 1997 APRIL 14-25

The Laguna Verde NPP has two units each with a BWR reactor with power output of 650 MWe. Unit 1 has been in operation since 1990 and Unit 2 since 1995. The mission was over a two-week span. It was carried out by two external experts for the IAEA plus three fire protection specialists from the Mexico Petroleum Institute. This mission was unique in that the scope was a fire safety audit required by the Mexican regulatory which adopted closely practice of the US Nuclear Regulatory Commission. It was accomplished with review of documents, discussions with counterparts, field inspection and evaluation of a fire drill.

The documents reviewed included: the overall fire protection programme, various fire prevention procedures, fire protection system testing and maintenance procedures and records, fire incident reports, previous reports of fire inspections and audits and follow-up actions, training records and the fire hazard analysis. Plant inspection was made to observe the installed fire protection systems and barriers , housekeeping practice and fire loads. An unannounced fire drill was witnessed by the Team.

Overall, the plant had a very well established and implemented fire protection programme. Only two audit findings were made:

1) The low level radwaste area in Unit 1 was not adequately protected and automatic sprinkler protection was recommended.

2) A compressed gas cylinder station for use in the laboratories and several locker cabinets were installed in the stairway in Unit 1 Control Building. Relocation of these items were recommended.

Housekeeping was good, except for minor deficiencies such as unsecured compressed gas bottles. Fire protection systems were adequate. Some discussions were made on the need for discharge relief valves on the fire water pumps. The Team also suggested the replacement of the deluge system in the Auxiliary Boiler House with a wet pipe sprinkler system to reduce potential for water damage. The Fire Hazard Analysis required some minor revisions. The fire drill demonstrated excellent response by the fire brigade.

8. SUMMARY

It is difficult to summarise these missions which covered plants of different designs and age, and which were reviewed with different scopes. The obvious remark is that no plant had perfect fire protection. However, in all cases, plant management and the regulatory authorities displayed a keen interest in fire safety, which was a reason why the missions were requested in the first place. It can be observed that each NPP must follow its own path towards improving fire protection, reflecting its own strengths and weaknesses, financial ability and regulatory tradition. This demonstrates the importance of communication between plants and countries in order to share experience and view points. The fire safety missions all proved useful to the host countries, not just in the results expressed in the formal documents but also in the dialogues that went between the IAEA Team and their counterpart. Te recommendations were offered only as advice to plant management. The most common ones include establishing a more formal fire protection programme and the position of a plant fire protection engineer, improvements in fire hazards analysis, and improvements in fire barriers.

The external experts that participated in the missions also benefited greatly, both professionally and personally. These initial contacts have paved the way for further exchange of technical information between the experts and the counterparts.

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FIRE SAFETY ANALYSIS: METHODOLOGY

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FIRE SAFETY ANALYSIS: METHODOLOGY

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1. INTRODUCTION

Should we be concerned about internal fires in nuclear power plants? Can we estimate the fire risk using quantitative methods? Are current state-of-the-art fire risk analysis methodology and practices adequate for decision making? Many believe that, given the changes that have been implemented in the older plants and the strict physical separations and fire protection equipment and practices that are incorporated in the modern plants, the issue of internal fires has been properly addressed for nuclear power plants, and the risk is sufficiently small. Much doubt is often expressed in our ability to model the fire phenomenon and thus in our ability to quantitate the risk of fires in a nuclear power plant. Thus, there are many who believe that quantitative fire risk analysis does not provide the proper basis for making decisions on fire related issues. This is perhaps best manifested in the fact that often the influence of elaborate and complicated fire protection systems is not clearly demonstrated in a fire risk analysis results.

I believe that fire risk is significant, it can be quantified and the quantified risk can be used for making decisions. This article addresses these three issues.

2. RISK SIGNIFICANCE OF FIRES

History and past risk studies [1] tell us that fire can be a significant contributor to the overall risk of a nuclear power plant. A number of fires have occurred that have had severe impact, and a few of those caused the failure of a large number of safety systems. Because of those experiences, we have done much to protect the nuclear power plants from fires. We have improved the quality of the materials, incorporated strict physical separation and enhanced the fire detection and suppression capabilities. Thus, we have reduced the likelihood of ignition and spread of a fire, increased the likelihood of discovery and mitigation, and reduced it for a fire to fail a minimal cut set of equipment.

The core damage frequency from fires estimated in fire risk studies for a number of plants within the USA ranges between 10^{-6} and 10^{-4} per reactor year [1]. Thus, although the overall plant risk may be considered as acceptable, for a large number of the plants, fire is a significant fraction of the overall core damage frequency.

3. RISK SIGNIFICANCE OF FIRES IN THE MODERN PLANTS

There is no doubt that because of all the defenses that are now incorporated into the design of modern power plants, the fire risk in these plants is less than the older plants. However, in my opinion, we still need to be concerned about internal fires for the modern plants. This is simply because an important part of the defenses against the effects of a fire depends on active components and humans.

The central control room contains a portion of practically all the important control and instrumentation circuits of the plant. Thus, a control room fire can potentially affect a large number of these circuits at the same time. In such an event, the fire risk is influenced by operators' ability to use the remote shutdown panels and by the ability of isolate that portion of the control and instrumentation circuits that are inside the control room. The latter is sometimes not a straightforward application of a switch. An irrecoverable failure (e.g., valve gets jammed into its seat) may occur from the fire affecting the control circuits inside the control room before the control room portion of the circuit is isolated.

In the case of modern plants and older plants as well, a fire outside the control room, that affects control and instrumentation circuits, may lead to wrong and confusing information on the control board. Current fire risk studies have not addressed such scenarios. The potential severity of such scenarios is not known. We do not know enough about the possible errors of commission under such conditions.

In addition to the control room fires, short term effects of smoke and the failure modes of new equipment types may also be important influences on the fire risk of modern plants. New equipment (e.g. electronic and computerized systems, distributed control systems, and data highways) is being introduced into the plants. The behavior of these equipment has not been completely understood under fire conditions. Recent tests have demonstrated that contrary to what was believed previously, smoke can have short term effects on electronic equipment. Many modern plants do not have perfect separation between redundant trains. For example, fire dampers have to close to establish the separation. One can postulate a scenario where sufficient amount of smoke escapes through a normally open damper (before it closes) and affects an electronic cabinet in an adjacent compartment.

In summary, even for the most modern plant, the fire risk may be sufficiently significant to deserve detailed scrutiny. Detailed analysis needs to be done to insure that unusual, albeit unlikely, phenomena are considered properly. There is a relevant analogy here with the common cause failure phenomenon; that is, the reliability of a highly redundant system is limited by the common cause element.

4. QUANTITATIVE ESTIMATION OF FIRE RISK

The risk of all types of hazards, which include fires in nuclear power plants, can be characterized by a quantitative measure using the probabilistic risk assessment definition and approach [2]. This can be done regardless of our level of knowledge about the phenomenon. The differences in our level of knowledge in various hazardous phenomena affects our level of uncertainty in the risk measures. We always know something about a phenomenon. I cannot think of any examples where we have total ignorance. The opposite is also true. We never have perfect knowledge. We always have some uncertainty. In summary, probabilistic risk assessment can be used in all types of situations and it will always include some uncertainty.

The methodology for fire risk analysis has evolved over the last 18 years [3-5] and has been scrutinized extensively through its application in numerous fire risk studies [1] and through comparative studies [6, 7]. Therefore, the question is actually not about our capability of quantifying fire risk. The question should be whether the level of uncertainties are such that the fire risk quantified per current practices can be useful for decision making.

Uncertainties in fire risk is discussed in some detail in Ref. [8]. A summary is provided below.

5. UNCERTAINTIES IN FIRE RISK

Fire is a complex phenomenon. Its complexity comes not only from the ignition and combustion processes, but from its impact on equipment and on the response of the operators to those failures. The fire fighting activity adds to this complexity through its own unique effects. Uncertainties in the fire risk are directly related to how well we can model the different aspects of the fire phenomena .

We can start from the very first element of a fire event, that is ignition of a fire. The uncertainties in this parameter have two elements: statistical and its correlation with later stages of the analysis. Statistical uncertainties are simply a function of the statistical evidence and experience base. The analysis of statistical uncertainty is well established. In the nuclear industry, the Bayesian approach is used for this purpose. As the experience base or statistical evidence increases the uncertainties in the parameter values decrease.

There is also uncertainty in the correlation between the fire ignition process with other elements of the fire risk model. The fire events included in the statistical evidence must include a threshold severity level, otherwise it would not be reported (e.g., sparks from an electrical connector or a smoldering cigarette in a waste basket). This threshold is certainly poorly defined because a spectrum of severities can be found in the incident descriptions. For example, there is at least one incident that was a smoldering fire and there are many that included a switchgear fire. When fire propagation modeling is considered the initial severity level of the ignited fire needs to be taken into account. Currently the threshold severity is not established and therefore this mismatch is a source of uncertainty in fire risk analysis.

The uncertainties in fire propagation models are deemed to be excessive and often it is stated that such models are completely erroneous. Both concerns are not well founded. Although the existing fire propagation models do not include all possible thermodynamic, heat transfer and chemical reaction aspects of a fire event, Refs. [9, 10] demonstrate that fire propagation models are capable of predicting the propagation patterns for a compartment within a reasonable uncertainty level. A part of the uncertainty in fire propagation modeling is associated with the input variables. A range of possible ignition source locations and characteristics, and other variables need to be simulated. Only a few of the simulated fire ignition scenarios match actual fire events that have occurred and are considered in the statistical analysis. For example, a compartment may include two vertically stacked cable trays that are separated horizontally. The fire risk analyst must consider a series of possible fire ignition scenarios in this compartment and analyze their propagation patterns. A number of these ignition scenarios, that may include extreme yet possible conditions, would cause damage to a critical set of the cables and the rest would not. For those scenarios that can cause damage, the analyst needs to assign an occurrence frequency. The statistical evidence (i.e., the fires that have already occurred in the nuclear power plants) may not include such ignition scenarios. Thus, the challenge becomes to estimate these frequencies in the face of little statistical evidence, which will clearly lead to large uncertainties. The methods for quantifying such uncertainties using the Bayes Theorem are well established.

There are many other sources of uncertainty in fire risk analysis that are perhaps of lesser importance than those discussed above. They include such issues as the ruggedness of various equipment and electrical cables to damage from fire and their failure modes. Current fire risk studies have often assumed the active fire barriers as highly reliable, and very few have considered situations where active or even passive fire barriers may get overwhelmed and fail to protect the opposite safety train. Short term effects of smoke are another source of uncertainty. Recent tests have demonstrated that aside from its obstructive effects, electronic circuits are susceptible to smoke in the short term. There is a large variety of styles in modeling control room fires and their effects on the operators. The same can be said about the cable spreading room as well. Other sources of uncertainty include models for fire detection and suppression that should be coupled with fire propagation analysis. The effects of suppressing medium (e.g., water or CO_2) on equipment not affected by the fire (e.g., from misdirected water stream) are often not included in the fire risk analysis.

Overall, in my opinion, we have relatively good understanding of the key elements of a fire event and, if the proper data and methodology are employed, the final results of the fire risk analysis would not entail excessive uncertainties. However, there is no doubt that there is much room for improving our level of knowledge.

6. FIRE RISK AND DECISION MAKING

A probabilistic risk assessment provides a ranking of the contributors to the risk. The contributors are generally accident scenarios that start from a perturbation in the balance of the plant and lead to an adverse consequence. In the case of fire risk analysis, the ranking is done in terms of fire scenarios. Fire scenarios include ignition of a fire at a certain location involving a certain combustible with clearly defined dimensions and characteristics. The fire scenario also includes fire propagation pattern, equipment exposure to the fire, cables and circuit failures, equipment failure as result of these exposures, automatic systems and operator response to these failures, fire detection and suppression activities, failure of other equipment that is not affected by the fire, and finally core damage. It may be noted that fire will eventually get extinguished, which is immaterial if equipment damage occurs prior to extinguishment.

The main purpose of performing a probabilistic risk assessment is to support a decision. There are three types of situations that a decision maker faces: either the risk is acceptable, the uncertainties in the risk are sufficiently small and the risk is unacceptable, or the uncertainties in the risk are large. The decision maker can reach a decision in all cases.

Risk acceptability depends on the criteria set forth by a regulatory agency or by other established practices. Overall plant risk has to be examined to determine whether the risk is acceptable or not. Fire risk may be a significant contributor to the overall plant risk, and yet remain acceptable because of the acceptability of the overall plant risk. If the uncertainties are sufficiently small and the risk is unacceptable, the ranking of various fire scenarios can be used to identify and prioritize possible plant modifications to reduce the risk. The modifications may be in terms of changes to the hardware or to the administrative elements or both.

Decision making in the face of large uncertainties may not be straightforward. Making changes to the plant when uncertainties are large may not be prudent. The ranking of various contributors that involve hardware elements may be masked. For example, the final risk analysis results may not be able to demonstrate the risk benefits of a sophisticated automatic fire suppression system versus a less sophisticated one.

The simplest approach may be a re-analysis of those elements of the fire risk that have the greatest influence on the uncertainties. This may not be a simple task, and may require a major research effort. For example, fire propagation analysis may be the main contributor to the uncertainties or a fire scenario may include a condition for which little or no statistical evidence is available. Since, fire risk is the result of combining a multitude of parameters, the lack of change in the uncertainty of a single parameter will not render the fire risk analysis results useless. Other parameters may be used to achieve a reduction in risk and even in the uncertainties.

7. CONCLUSION

From a review of the fires that have occurred in nuclear power plants and the results of fire risk studies that have been completed over the last 17 years, we can conclude that internal fires in nuclear power plants can be an important contributor to plant risk. Methods and data are available to quantify the fire risk. These methods and data have been subjected to a series of reviews and detailed scrutiny and have been applied to a large number of plants. There is no doubt that we do not know everything about fire and its impact on a nuclear power plants. However, this lack of knowledge or uncertainty can be quantified and can be used in the decision making process. In other words, the methods entail uncertainties and limitations that are not insurmountable and there is little or no basis for the results of a fire risk analysis fail to support a decision process.

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Invited Paper

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FIRE RISK ANALYSIS: A DISCUSSION ON UNCERTAINTIES AND LIMITATIONS

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Abstract

A fire risk analysis, using probabilistic methods, attempts to model fire scenarios that can be described in terms of the following elements: ignition of fire, growth of the fire, detection and suppression processes, impact on cables and other equipment, and response of the automatic safety and control systems and plant operators. The level of uncertainties and limitations of the analysis varies among these elements. Although the potential failure modes of cables and electrical circuits have been debated for a long time, there are some uncertainties in our understanding of the failure modes of equipment under fire conditions. Smoke propagation and smoke damage have generally been omitted from fire risk studies. Shorts within electronic circuit boards caused by soot deposits are not modeled, such failures can have an impact on the information provided to the control room operator. The operators' response to the changes on the control board is certainly a complex issue. The behavior of control room operators when there is a fire in the control room is also the subject of much debate. Of specific concern is the proper transfer of the controls to the remote shutdown panel. Another area of much debate centers around the control of combustibles. Several studies have taken credit for the house-keeping procedures to screen potential fire scenarios in areas of a plant that contain a large collection of cables (e.g., a cable tunnel or cable shaft). In such cases, clearly the debate is over the likelihood of a fire that can cause damage. None of the fire risk analyses familiar to the authors have properly modeled the simultaneous effect of a fire on multi-units.

1. INTRODUCTION

A large number of fire Probabilistic Risk Assessments (PRAs) and Independent Plant Examination for External Events (IPEEEs for U.S. plants) have been completed for nuclear power plants. These fire risk studies have addressed many risk significant issues and have shown that fire risk can be an important contributor to overall plant risk [1]. Yet, these studies have also shown that fire risk is much reduced because of the separation among redundant trains, fire protection features, and control of combustibles and ignition sources. The reduction in fire risk leads us to question whether the analysis has given proper consideration to those aspects of a fire event that have either been modeled simplistically or not addressed at all because of their low likelihood of occurrence.

The uncertainties and limitations of fire risk studies are first discussed in terms of various aspects of a fire event. Further discussions are provided for those aspects that in the opinion of the authors have not received sufficient consideration in current fire risk studies. It may be noted that some of the phenomena discussed in this article may be irrelevant or extremely unlikely for most locations and situations in a nuclear power plant. However, this statement cannot be generalized even within one plant. Most plants include some unique settings that give rise to a complex chain of events that is not typically addressed in detail in fire PRAs.

2. SOURCES OF UNCERTAINTY AND LIMITATIONS

The following is a list of issues and phenomena that influence the uncertainties and limitations of the final results of a fire PRA:

- Fire ignition frequency - Many PRAs have used the Licensee Event Reports (LERs) as the basis for establishing the fire ignition frequency. It is argued that the actual number of fire events could be as much as 10 times higher than that reported. On the other hand, it can also be argued that those fires that did not get reported were not significant enough. Although, the uncertainties in the overall frequency of fire occurrence may be small, the uncertainties in the frequency for a specific location and for specific conditions are large. Many PRAs have used various factors to adjust the overall frequency for the location and severity of the fire. An important source of uncertainty for the fire frequency lies in the correlation between the initial fire considered in fire propagation analysis and the severity of the fire modeled by the fire frequency.
- Fire propagation - The main purpose of fire propagation modeling is to demonstrate whether the fire can propagate and damage the equipment and cables that are of concern. Also, if a damage possibility exists, the probability of damage is estimated from a comparison of the timing between propagation, detection and suppression. There is much debate on how well the existing programs model the actual fire propagation process. It is believed that the uncertainties in these models are the largest among all the parameters used in a fire risk analysis. Refs. [2-4] discuss the features and limitations of various fire propagation models that have been used in fire risk studies.

  Equipment damage - The most important item of concern in a fire incident is electrical cable damage. This is because a large number of electrical cables may be present in a small location and cables are susceptible to excessive heat. Electrical cabinets that contain a large number of switches are also found to be important in fire risk analyses. The parameters needed to model cable damage in a fire propagation analysis are discussed in this article as a potential source of uncertainty.
- Equipment failure modes - Much effort has been expended in deterministic fire hazard analyses to establish the failure modes of equipment and especially electrical circuits (as a results of cable failure). Although the level of uncertainty in this aspect of a fire risk study should be minimal, many risk studies have used probabilistic arguments (although not explicitly quantified) to ignore the potential for some of the severe events (e.g., fire induced LOCA) by stating that either further damage to the cables would reverse the adverse condition or the specific chain of events is extremely unlikely. This issue is further discussed in this article.
- Detection and suppression - Fire detection and suppression analysis is an integral element of fire propagation analysis. The timing between the two phenomena is

compared to establish the probability of damage before suppression. Although little experimental data are available to verify the probabilities obtained from this analysis, since typically failure probabilities are large, the uncertainties compared to other parameters of a fire scenario are small.

- Control room fires - Many fire risk studies have identified the control room as a significant risk contributor and meanwhile many have used a simplistic model to analyze such fire scenarios. Since the control room is practically the only location where all redundant circuits meet, and operators can be directly influenced by the effects of a fire, a simplistic model may not be sufficient. Control room fire models have been further discussed in this article.
- Cable spreading room fires - Similar to the control room, the cable spreading room may contain a large number of circuits from the redundant train. Often, risk studies have discounted such areas based on the presence of qualified cables and strict housekeeping practices. Since the area contains a large number of redundant circuits, the uncertainties in the risk impact should be quite large.
- Barrier failure - Two types of barriers are present: active and passive. Under special circumstances, a passive barrier may be overwhelmed by a severe fire. Such scenarios are rare for nuclear power plants and can only be envisioned for turbine building fires. Active barriers may fail to close properly and allow for smoke and other gases to pass through to adjacent compartments. Such scenarios are typically considered very unlikely. Much uncertainty exists in active barrier failure probability, especially when systems interaction (e.g., effect of air flow while the ventilation system is running, Ref. [5]) is considered. Since barrier failure can affect redundant trains, these uncertainties may be significant.
- Smoke propagation and impact - Smoke impact on equipment is typically considered as a slow process and therefore of little importance to fire risk analysis. Recent tests have demonstrated that short term failures are possible and therefore it may be necessary to add smoke impact to fire risk studies. This issue is further discussed in this article.
- Multi-unit analysis - Several multi-unit plants include compartments that contain equipment and cables from both trains. Rarely has a fire risk study addressed the impact of a single fire on both units at the same time. Of course, for the majority of the cases, only one unit is severely affected. This issue is a limitation that can easily be overcome. However, it raises the question whether multi-unit core damage risk should be treated differently.

3. CABLE DAMAGE THRESHOLDS

Typically, in fire propagation analyses, a single temperature damage threshold is assumed for the cables of interest, and simplistic models of the fire behavior and heating of the target cables are applied. If the predicted cable response temperature reaches the threshold then failure is assumed. The damage limit for modern cable insulation materials is typically assumed to be on the order of 350°C, although the authors have noted use of values ranging from 150 to 450°C. There is nothing inherently wrong with the concept of a thermal damage threshold for a cable application. However, there are some aspects of cable failure phenomena that are often neglected in the simplistic treatment.

First, it should be noted that the threshold values used in modern assessments are typically based on tests performed by Sandia National Laboratories (SNL, Refs. [6, 7]), other potential sources of data having been largely discredited [8]. The SNL tests were performed using an oven-type thermal exposure in which the time to catastrophic failure (dead shorts between conductors) versus exposure temperature is measured. The threshold values are typically taken as that temperature resulting in failure times in excess of one hour. The SNL tests involved an imposed voltage potential without any imposed current. The tests were designed primarily to assess the performance of light power and control cables, and only three cable types have been tested directly for fire performance.

Uncertainty is introduced because (1) the cited cable performance tests are quite limited in scope, and yet they are being widely extrapolated as representative of (or conservative bounds for) all cable types and circuit applications, and (2) because the risk analysts have not directly accounted for the critical factors that will influence both cable and circuit performance. The current practice represents an often under-recognized source of analysis uncertainty. That is, analysts uniformly recognize that the cable damage threshold is an uncertain parameter, but typically fail to appreciate the various factors that contribute to this uncertainty. Further, in the view of the authors, this source of uncertainty can be reduced significantly with only limited additional effort. Points not adequately accounted for nor recognized in typical risk assessments include the following:

- Tests to assess the performance of cables in the context of severe accident equipment qualification (EQ) have extensively explored the thermal behavior of a wide range of electrical cables (e.g., see Ref. [9]). EQ test results performed in a superheated steam environment have been shown to correlate well with the fire exposure tests [10], but the EQ tests provide a much more complete picture of cable electrical performance for a much broader range of cable types and products. These data have not yet found wide application to fire risk assessment.
- The broad base of EQ tests, and even the limited base of direct fire exposure tests, illustrate that cable performance is a strong function of the cable insulation materials. There is a wide range of materials used in modern cable construction. The commonly applied threshold value of 350°C corresponds to gross failure of a typical cross-linked polyethylene material. Other materials may display either a lower (e.g., non-cross linked polyethylene or poly-vinyl-chloride) or higher (e.g. silicone) threshold of damage. In many cases it may be impractical to verify the cable material for a given case, but in others verification may be quite easily accomplished. In particular, the original design specifications and cable sizing documents can be consulted without need for field verification. In the U.S., a typical plant ampacity verification study, now quite commonly available, will provide detailed information on both the cable physical characteristics and on the cable ampacity loads for each safety-critical cable.
- The critical measure of a cable's electrical performance is insulation resistance (IR). Cable IR has been found to decrease with increasing temperature in a relatively well behaved and readily predictable manner (typical behavior is roughly linear when plotted on a log-log scale). Data is available to characterize this behavior for a wide range of cable insulation materials. Hence, the performance of cables in an elevated temperature environment is readily amenable to analysis, and yet no fire risk assessment to date has taken advantage of this fact.
- The authors are unaware of any fire risk assessment performed to date that has fully assessed cable performance requirements. For a risk assessment, treatment of a given cable circuit fault as a threshold behavior is appropriate. However, the damage threshold assumed should be based on a more reasoned understanding of cable IR behavior and circuit performance requirements rather than on blind application of a single threshold

value based on references to past practice. For example, a 4-20 mA instrument circuit will be far more sensitive to cable IR degradation than would a typical power or control circuit. Hence, the failure analysis of an instrument cable should include a more stringent failure threshold than that applied to a power or control cable. The performance requirements and fault tolerance of each circuit under analysis, or at the very least each class of circuits (i.e., instrument, power, and control), should be considered individually.

- It is well known that an energized cable will experience a self-heating effect due to the flow of current and the internal resistance of the conductors. This behavior is especially important for normally energized power cables. Under current U.S. design practices [11, 12], an ambient temperature of 40°C is typically assumed, and electrical currents (or ampacity) is limited to ensure that the conductor temperature will not exceed 90°C. This implies that the self-heating temperature rise for a power cable may be as high as 50°C above the prevailing ambient. Even if the particular cable of interest is not itself normally energized, it may be co-located with other normally energized power cables in the same cable tray or conduit and may still be subject to substantial pre-heating. As noted above, the tests upon which cable damage thresholds are typically based included no imposed conductor currents; hence, they do not reflect any consideration of potential self-heating effects. A direct application of the identified damage thresholds without the consideration of the added impact of cable self-heating may substantially overstate the performance limits of normally energized power cables. This behavior is also readily amenable to analysis using existing methods.

In summary, the current treatment of cable damage behavior has not taken adequate advantage of the current knowledge base regarding cable performance requirements and limits. There is a considerable under-utilized base of experimental data on cable performance that could reduce the direct parameter uncertainty associated with cable damage thresholds for a given product. Further, there is also a lack of adequate appreciation in fire risk assessments for the factors which will determine the actual performance limits of a given cable in a given circuit application.

As a general observation, the current practice is effectively treating all cables as if they were light power or control cables, insulated with a cross-linked polyethylene material, and only called upon to perform a short-duration function in response to a fire event. For specific risk scenarios in which this is representative of the actual function of interest, the current treatment is probably about as good as one can expect to achieve given current knowledge. However, for the wide range of other applications including other cable types, instrumentation circuits, or any normally energized, longer-term demand, or high voltage power circuit, the current practice contributes to an additional source of uncertainty that is both largely unrecognized and unnecessary.

4. FAILURE MODES OF ELECTRICAL CIRCUITS

The chain of events for equipment damage can be extremely complex. There are many examples from either actual events in nuclear power plants or failure modes discovered in the process of a safety review that indicate that, despite the long history, new failure mechanisms are being discovered. Of specific importance is the impact of a fire on cables and the potential failure modes of electrical circuits. These failures modes have been debated since the early times of fire risk analysis [13]. Several NRC Generic Letters [14-17] and other publications [18] provide examples of how such failures can occur.

Of primary importance is the failure modes of control circuits either from direct exposure to fire conditions or as a result of a failure in their associated cables. A cable may fail in one of the following ways – intra-cable wire to wire contact, wire to ground contact, wire to an active source contact (hot short), inter-cable wire to wire contact and severed wire. The first two failure modes are deemed to be the most likely. It is thought that hot shorts have been observed in fire incidents. As a result, a circuit may experience shorts within and without the circuit (the hot short), short to ground and severed wiring. Depending on which wires within a circuit experience these failures, the controlled item will behave accordingly. The following is a sampling of failures that have either been experienced or identified during circuit analysis:

- All permissive signals are bypassed and a normally closed valve opens spuriously.
- Power operated safety valve cycles open and closed.
- Valve motor gets energized while the torque switch is bypassed and valve gets jammed into the seat.
- Two valves receive signal to open simultaneously as a result of one cable failure.

In addition to cables, electronic circuit failures associated with the control circuitry can induce similar failures. For example, a short in a transistor or a short across various conductors caused by soot deposits may simulate wire to wire contact. The effects of soot and smoke are discussed separately below.

Often detailed deterministic analysis is conducted for the components included in the PRA model for the potential for spurious actuation. In such studies individual circuits, associated cables and their respective locations are analyzed for possible shorts among wires. To minimize the possibility of missing a failure mode, a systematic method such as fault tree analysis may be needed. The likelihood of these shorts is estimated by simply considering the number of wire contact possibilities within the cable [13]. Although the uncertainty in the likelihood of a short is rather small (because values greater than 0.05 are used), much uncertainty may exist in the completeness of the list of failures.

The methodology for identifying such failures is based on the internal events PRA. Many plants have used the safe shutdown analysis done as part of their Appendix R compliance effort for this purpose. A thorough comparison between the PRA model and Appendix R safe shutdown components is needed in this case. The internal initiating events should be subjected to a fault tree type analysis to identify the component and from that the cables that can cause the initiating event. Loss of offsite power is one example, for which the power and control circuits should be identified and the associated cables traced. In addition to the differences in the plant impact models, further uncertainties are introduced when likelihoods are assigned to various failure modes. No experimental data exists to support these likelihoods. Given lack of sufficient knowledge, the conservative approach may be to assume the worst failure modes for the event sequence.

5. SYSTEMS INTERACTION

The above discussion is focused on individual circuits and their associated cables. In addition to the heating effects, failure can occur from smoke and suppression activities. Both phenomena can affect equipment outside the immediate fire area. The possibility and impact of smoke propagation is discussed in a separate section below. Water from suppression activities can travel to areas below the fire and affect electrical equipment. Incidents have occurred where water from the suppression system, because of plugged drainage piping, has found its way down to electrical cabinets and has caused shorts and equipment failure modes similar to those discussed in the preceding section. The cooling effects of CO_2 are also suspected of adversely affecting electronic and electromechanical devices.

The methodology that can be used to identify such scenarios may be based on an event tree, that starts from a fire in a specific location and questions the availability of various elements of the suppression activity and fire barriers, etc.

6. SMOKE PROPAGATION AND DAMAGE

Only in a few fire risk assessments has the possibility of smoke propagation been considered explicitly. However, in all cases, it is assumed that smoke, aside from being a nuisance to operators and fire fighters, has no short term effects. Equipment may have to be cleaned thoroughly or replaced because of exposure to smoke, but it is always assumed that such equipment will remain functional during a fire event.

In some limited sense, it can be argued that smoke damage is included in the initial screening steps of a fire risk assessment, where given a fire in a compartment it is assumed that the entire contents of that compartment is failed. This approach often fails to include the normally open passages between compartments that are protected by fire dampers, doors or other isolation devices that may fail to close before allowing some amount of smoke through. Furthermore, at almost any stage beyond the first screening steps smoke is ignored completely. In particular, one of the first assumptions relaxed will typically be the extent of the postulated fire damage. Fire models are applied to assess the potential for and timing of critical component damage based only on consideration of direct thermal damage. Further, this assumption is not applied uniformly to all fire areas. In particular, it is never applied to the main control room.

Clearly, smoke damage has long been a concern to fire safety specialists and the insurance industry. However, in the context of electrical equipment it has typically been assumed that smoke is only a problem in the longer term (days or even months after a fire).

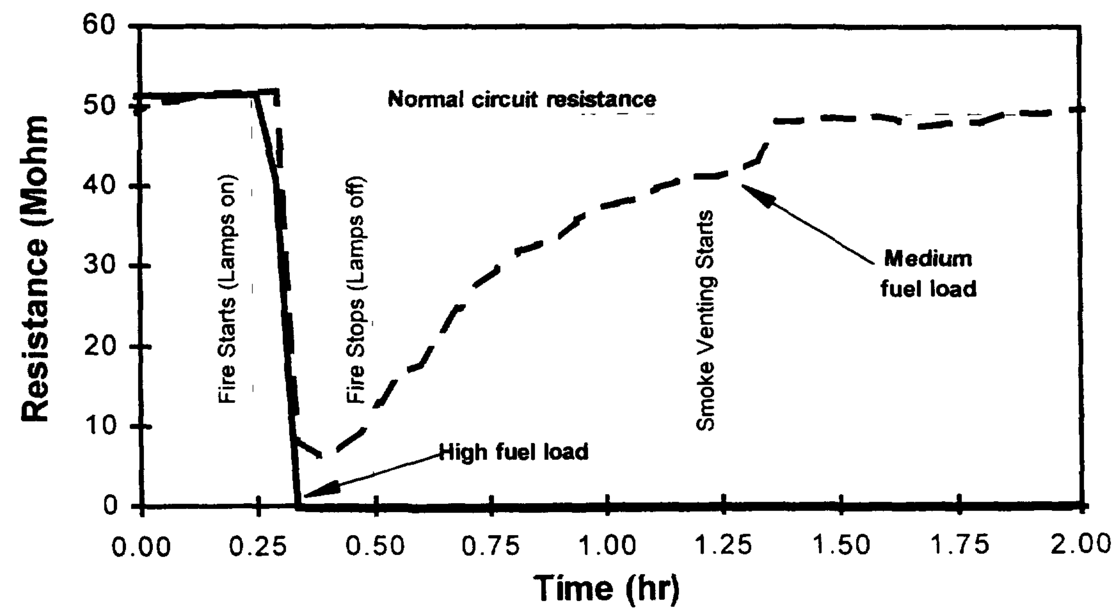

FIG. 1. Resistance measured across leads on a printed circuit under smoke conditions.

Smoke is not assumed to be a concern from the perspective of the short-term system failures of interest to a risk analysis. This is based on the assumption that the only significant impact of smoke on electrical equipment will be caused by long-term corrosive attack. Evidence has now developed to illustrate the fallacy of this myth.

For many years there have been anecdotal accounts of smoke having caused the direct and immediate failure of high voltage equipment. Actual documentation of such incidents has proven elusive however. In the aftermath of a fire event it is quite difficult to reconstruct the exact sequence of events leading to individual component failures. Indeed, for most applications, direct equipment damage is not of particular interest; hence, experiential data to support a short-term smoke-damage concern is lacking. However, recent test results Refs. [19, 20] clearly demonstrate that, at least for solid state electronics, short term smoke damage is a real problem. The tests also illustrate a potential for this same damage mechanism to impact high voltage electro-mechanical components as well (e.g., switchgear and circuit breakers) although no direct testing of these types of equipment has been undertaken.

The new tests have examined a range of both simple target objects, such as individual chips and circuit traces, and a selection of functional digital circuits. Monitoring of these targets is performed in real time concurrent with the smoke exposure. The results have illustrated that the exposure of such equipment to smoke results in an immediate (within seconds or minutes) and pronounced degradation in the circuit integrity. Further, once the smoke has been vented from the exposure chamber, much of the degradation is recovered. This behavior has led the researchers to conclude that deposition of soot onto the component is not required for degradation to occur. Rather, the mere presence of airborne smoke particulate is sufficient to cause significant degradation in circuit integrity. This has been attributed to ionized airborne smoke particles creating a path for current leakage.

Figure 1 illustrates the measured circuit resistance versus time response of a high-voltage, low-current printed circuit under both medium and high fuel (smoke) load exposure conditions. The normal circuit resistance is 50 Mega-ohms. Note the sharp drop in resistance within 1-4 minutes of the fire start time. This indicates a smoke-induced short-circuit pathway that bypasses the normal circuit resistance elements and severe circuit degradation. It is especially interesting to note that for the medium fuel load, once the smoke is vented from the exposure chamber, a full recovery of the original circuit integrity is realized. This clearly indicates that for this case, it is airborne smoke particulate that is responsible for the degradation. For the high fuel load case, no recovery is realized indicating the longer-term impact of soot deposition due to the higher smoke loading.

While the tests to date have been limited to digital components and circuits, the same mechanism could also impact higher voltage equipment by creating current leakage paths which might in turn lead to loss of an electrical bus through tripping of circuit protection devices. A new series of tests is planned to assess this behavior under higher voltage conditions, and a physical model of the degradation process may be forthcoming. For the purpose of the current discussion it is simply important to acknowledge that, contrary to popular opinion, smoke does have the potential to induce short-term component failures. While the threshold of such damage remains uncertain, the potential has been demonstrated. As a future area of development, it will be important that the methods of risk assessment be expanded to include smoke damage. Until such time, this will remain an area of uncertainty in our fire risk analyses.

7. CONTROL ROOM FIRE MODELING

Control room operators can be affected by the fire in two ways. A fire may occur inside the control room that may force the operators to leave the room or it may occur at a location where it can adversely affect parameters that are displayed on the control board and lead to operator confusion. The first scenario is addressed routinely in fire risk studies and many have demonstrated that the control room is an important contributor to fire risk. The second scenario is rarely addressed. Two fundamental issues need to be addressed - can the control room operators make the situation worse than what it is and can the plant be safely shutdown from a location outside the control room under all possible fire conditions?

A majority of fire risk studies have used a simplistic methodology and conservative probability values for this purpose. A conservative methodology may be sufficient in most cases if further risk reduction is not desired. However, many valuable lessons can be learned from a thorough analysis of a fire event inside the control room. Depending on which panel is affected by the fire, the plant will respond differently and therefore a different mode of operation may be required from areas outside the control room. A thorough analysis of the circuits that may be affected by a fire in the control room may be needed to ascertain that the operators will have sufficient time to apply the transfer switches and isolate the control room and activate remote shutdown capability. Such circuit analysis may follow the same fault tree type methodology as for control circuit failure mode analysis. An event tree model for the behavior of the control room operator may be developed in terms of possible internal event initiating event (will depend on the location of the fire in the control room) and actions that the operators have to take to reach safe shutdown. The analysis may also include the timing for control room abandonment, a major source of uncertainty in the analysis. When questioned, plant personnel often state an intent to remain in the main control room as long as it is possible. This may actually lead to further problems if the main control room functions are, in fact, rendered inoperable. Thus, if the model assumes a timely abandonment, may not reflect realistic conditions.

8. CONCLUSIONS

The level of uncertainties and limitations of the analysis varies among the fundamental elements of a fire event. The level of uncertainty attributed to fire propagation and damage analysis is deemed to be large. An important element of that analysis is the threshold temperature for cable damage. New data are available to support the proper modeling of cable failure that have seldom been used by fire risk practitioners. Although the potential failure modes of cables and electrical circuits have been debated for a long time, there are some uncertainties in our understanding of the failure modes of equipment under fire conditions. Smoke propagation and smoke damage have generally been omitted from fire risk studies. Shorts within electronic circuit boards caused by soot deposits are not modeled, such failures can have an impact on the information provided to the control room operator. The operators' response to the changes on the control board is certainly a complex issue. The behavior of control room operators when there is a fire in the control room is also the subject of much debate. Of specific concern is the proper transfer of the controls to the remote shutdown panel.

The uncertainties and limitations of fire risk studies are found to be sufficiently significant. Every power plant has at least a few unique features that make some aspects of the above discussed issues to be important for consideration in the fire risk assessment. In other words. most plants include some unique settings that give rise to a complex chain of events that is not typically addressed in detail in a typical fire PRA.

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RESEARCH NEEDS IN FIRE RISK ASSESSMENT

(Summary)

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The objective of this paper is to discuss a number of areas where improvements are needed in fire risk analysis methods and data to: a) provide a better understanding of the risk contribution due to fires in nuclear power plants, and b) support improved decision making regarding nuclear power plant fire protection. It is expected that the issues presented in this paper will be integrated with previously identified fire protection issues (some of which are based on conditional risk considerations) as the agency develops a fire research program.

While there is little argument about the potential importance of fires, the magnitude of the fire risk and the specific measures needed to efficiently manage this risk are not as clear when considering individual plants. This uncertainty is due to uncertainties in the current state of knowledge concerning the initiation, growth, suppression, and plant impacts of fire-induced nuclear power plant accident scenarios. These latter uncertainties are reflected by the variability in methods and data used by current fire risk assessments (which contribute significantly to variations in predicted fire risk magnitudes and profiles), and by the ongoing dialog between the U.S. Nuclear Regulatory Commission (NRC) and industry regarding the usefulness of current fire risk assessment tools in supporting proposed plant changes and the development of a risk-informed, performance-based rule for nuclear power plant fire protection.

Improvements in the NRC staff's ability to thoroughly understand and accurately evaluate nuclear power plant fire risk require efforts in three areas: fundamental research on material properties and scenario phenomenology, the development of methods and tools to apply the results of fundamental research, and the application of these methods and tools to actual plants. It is anticipated that improvements sufficient to ensure good decision making also require efforts in these areas, although probably to a lesser degree. Clearly, the amount of effort required depends on the set of decisions to be made.

An examination of the fire risk analysis process has been performed to systematically identify areas where additional research may be needed. This examination considers the treatment of fire initiation, fire scenario-induced equipment damage (which involves the competition between fire growth and suppression processes), and plant response to the loss of equipment. The results of this examination have been supplemented with input from a variety of sources, including the NRC's Fire Protection Task Action Plan, insights gained from the NRC's review of recent Individual Plant Examination of External Events (IPEEE) studies, a number of recent relevant papers and reports, and comments provided by the NRC's Advisory Committee on Reactor Safeguards; the resulting list of issues is provided in Table 1. It can be seen that many of these issues are associated with uncertainties in material properties and fire phenomenology (e.g., the generation, transport, and deposition of smoke in nuclear power plant fires). Others, however, concern impacts on hardware (e.g., the likelihood and consequences of electrical hot shorts) and humans (e.g., confusion due to spurious indications).

The list of issues in Table I is preliminary. Following discussions within the NRC, it will be finalized and then prioritized. The prioritized list will, together with the results of discussions with industry and interested international organizations, provide key input into the development of NRC's fire research program.

Table I. Potential Fire Research Issues

Analysis	Task	Potential Issues
Fire Initiation	Fire frequency analysis	Rated vs unrated cables
		Effect of plant operations, compensatory measures
		Adequacy of database
		Transient-fueled fires
	Fire frequency partitioning	Likelihood of severe fires
		Frequency-magnitude relationship
Fire Environment	Single compartment analysis	Cable tray fires
		Electrical cabinet fires
		Large oil fires
		Hot gas layer development
		Flashover
	Multi-compartment analysis	Hot gas layer development
		Backdraft
		Flashover
	Smoke generation & transport	Smoke generation
		Smoke buildup
		Smoke transport
Fire Detection and Suppression	Detection analysis	Adequacy of data
		Scenario-specific analysis
	Suppression analysis	Automatic suppression effectiveness
		Manual suppression effectiveness
		Effect of compensatory measures
		Scenario-specific analysis
Fire Containment	Barrier reliability analysis	Adequacy of data
		Analysis tools
		Penetration seals

Table I. Potential Fire Research Issues (Continued)

Analysis	**Task**	**Potential Issues**
Hardware/Human Performance	Component fragility analysis	Circuit failure mode and likelihood
		Cable thermal fragilities
		Smoke fragility of electronic equipment
		Cabinet fragilities
		Suppressant-related fragilities
	Human reliability analysis	Cognitive impact
		Environmental impact
		Role of fire brigade in plant response
Integrated Fire Scenario		Main control room fires
		Turbine building fires
		Containment fires
		Circuit interactions
		Uncertainty analysis
		Multiple unit interactions
		Non-power and degraded conditions
		Scenario dynamics
		Seismic/fire interactions
		Flammable gas lines
Other		Learning from experience
		Comparison of methodologies
		Standardization of methods

IAEA-SM-345/25

FIRE PSA METHODOLOGY

XA9847507

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Abstract

The fire PSA methodology, which NUPEC has introduced from one of US IPEEE fire PSA methodologies, was applied to a Japanese typical 1,100 Mwe class four loop PWR to confirm the applicability. Through this application, some consideration is given on some key parameters, such as fire frequencies and severity factor, in the fire PSA methodology to develop the fire PSA models specific to Japanese plants.

1. INTRODUCTION

NUPEC has developed a fire PSA methodology since 1992 sponsored by Ministry of International Trade and Industries. After the introduction of one of US IPEEE fire PSA methodologies(1),(2) to NUPEC, we have applied the fire PSA methodology at full power operation to seven particular areas of a Japanese typical 1,100 MWe class four loop PWR plant to confirm the applicability of the methodology. This presents the main results of the trial application(3) and some consideration on the fire PSA methodology through the application.

2. FIRE PSA METHODOLOGY DEVELOPED IN NUPEC

The fire PSA methodology developed in NUPEC consists of three stages, namely, “spatial interaction analysis”, “screening analysis” and “detailed analysis”, as shown in Fig.1. Spatial interaction analysis makes fire scenarios based on plant information. Screening analysis identifies risk significant fire scenarios under conservative assumptions. Detailed scenario analysis makes sub-scenarios without the conservative assumptions and quantifies core damage frequencies for the sub-scenarios.

3. MAIN RESULTS OF TRIAL APPLICATION OF THE METHODOLOGY TO A JAPANESE PWR PLANT

In order to confirm the applicability of the methodology to the Japanese LWR plants, to develop Japanese specific fire PSA models and to investigate the way to reflect Japanese specific circumstances to the methodology, we applied the fire PSA methodology at full power operation to seven particular areas of a Japanese typical 1,100 MWe class four loop PWR plant, namely, a turbine building, a main transformer area, a switch-gear room for safety components, a diesel generator control room, a battery charging room, a component cooling water system (CCWS) pump area and a reactor cooling pump (RCP) area. The PWR studied is a 1,100 MWe class four loop type with a large dry pre-stressed concrete containment vessel (PCCV), which has, as the engineered safety features, high pressure injection systems(HPIS), low pressure injection systems (LPIS), containment spray systems, four accumulators and auxiliary feedwater system s(AFWS). The AFWS composes of three trains including two motor-driven pumps and one turbine-driven pump.

3.1. Spatial interaction analysis

First, fire areas are defined as the areas surrounded by 2- or 3-hour rated fire barriers, and fire zones are defined by subdividing fire area. In this study, however, we defined fire scenarios for each fire area without subdividing fire area into fire zones for lack of plant information and fourteen local

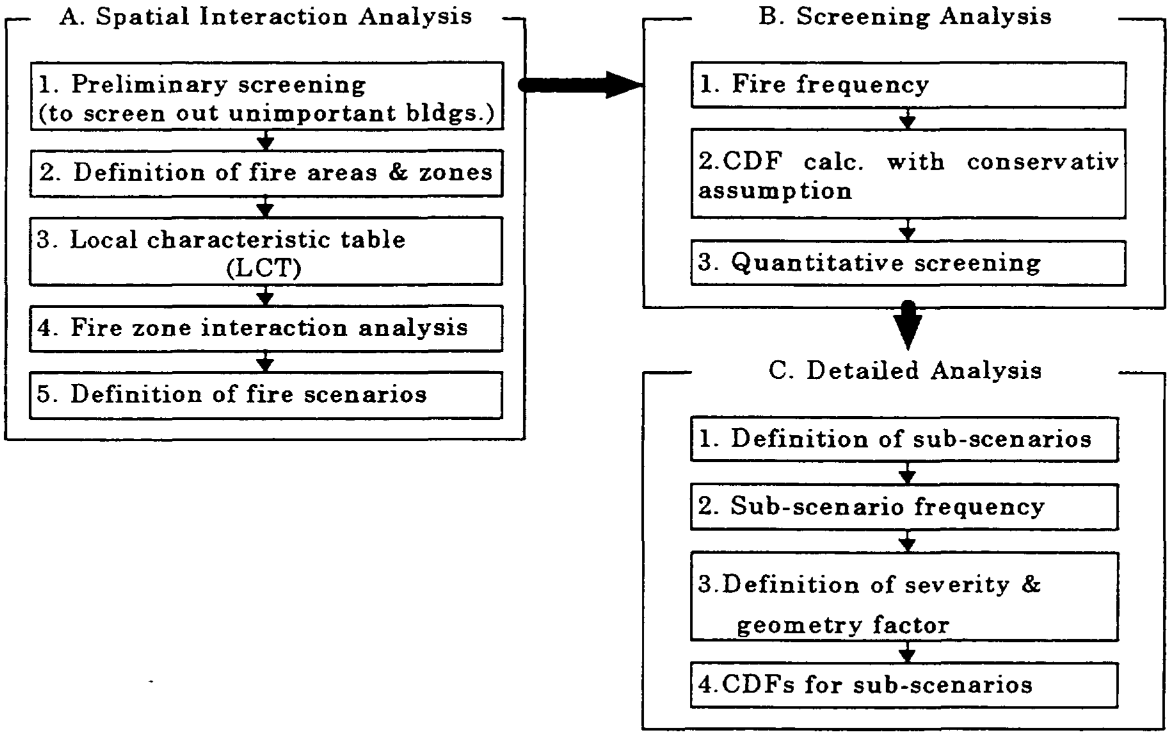


Fig.1. Fire PSA methodology developed in NUPEC.

characteristics tables (LCT) were prepared including fire propagated areas. LCT includes fire loads, fire protection features, safety related equipment and so on in the fire area. Using these LCTs, we made 7 local fire scenarios and 10 propagation scenarios for the seven particular areas.

3.2. Screening analysis

Core damage frequencies(CDF) for the above identified fire scenarios were quantified using the conservative assumption that all safety related equipment associated with the fire scenario was damaged. CDFs were defined as products of fire frequency of the fire area, conditional initiating event occurrence frequency, conditional CDF (and fire propagation probability for the propagated fire scenarios). Fire frequency was estimated considering the number of components as fire sources in the fire area. The Japanese fire experiences until March 1996 were used to estimate fire frequencies. We selected five fire induced initiating events based on internal event PSA. In addition, initiating events induced by mal-function of valves due to hot short in electric circuits were taken into account. The occurrence frequencies of the fire induced initiating events were obtained by system analyses. We made accident sequence analyses for the above selected fire-induced initiating events. Conditional CDFs for the fire scenarios were estimated by quantifying the event trees under the assumption that all safety related equipment associated with the fire scenarios was damaged. Quantitative screenings for detailed analyses were conducted to identify scenarios to be analyzed. The 4 of 17 fire scenarios were identified as the risk significant fire scenarios.

3.3. Detailed analysis

In order to remove the conservative assumption in the screening analysis, sub-scenarios were defined for each screened fire scenario, where concepts of both severity factor for equipment fire and geometry factor for transient fire were introduced. Both severity and geometry factors are a kind of attenuation factors from fire source to the target equipment. CDF for sub-scenario was defined as the products of sub-scenario frequency, conditional CDF, severity or geometry factor and non-

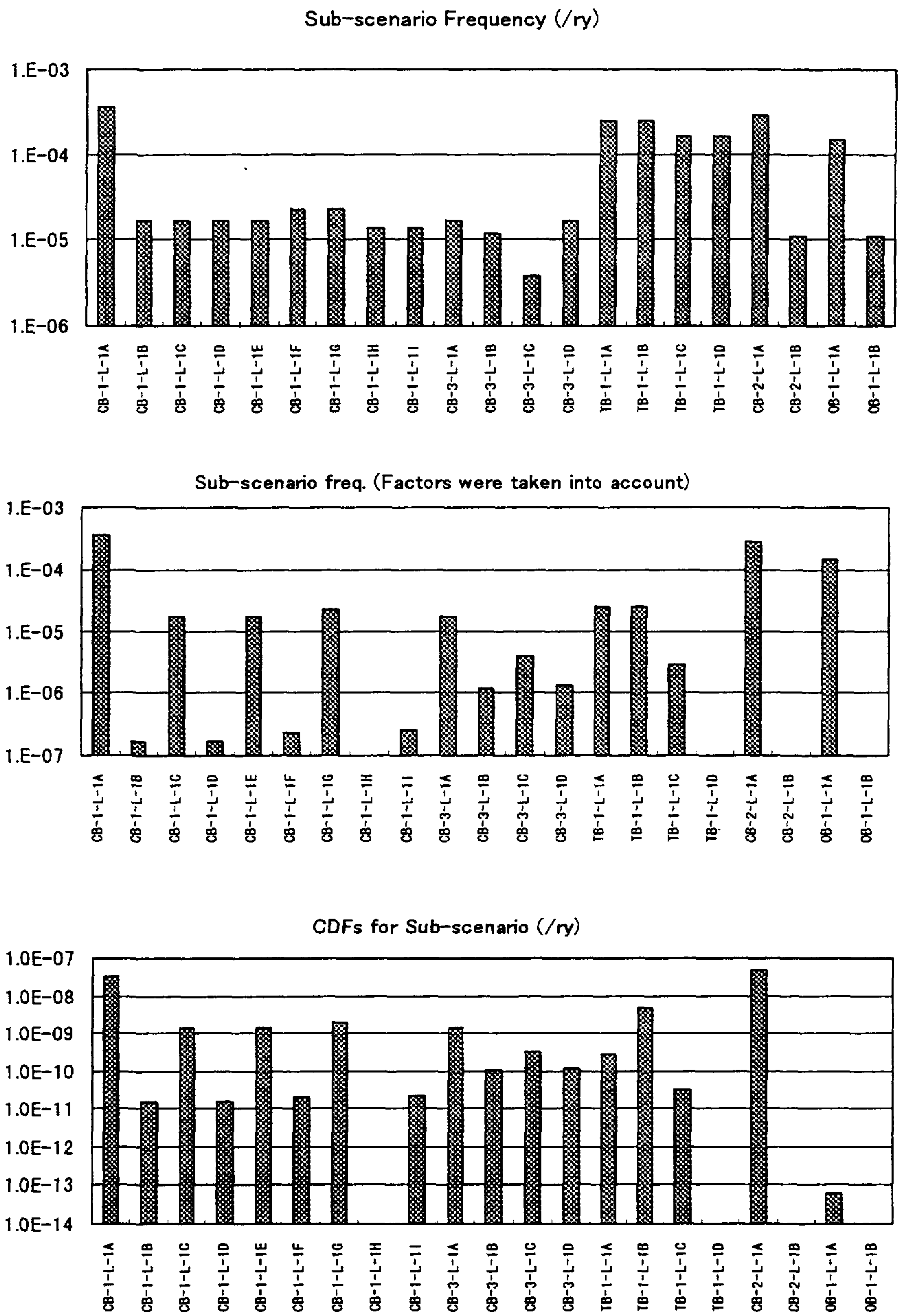


Fig.2. CDF for sub-scenarios.

suppression factor. In principal sub-scenarios are defined for each individual fire source. In this study, however, sub-scenarios were not always made for every individual fire source, but some fire sources of which impacts were the same were grouped into one sub-scenario. We made 21 sub-scenarios in all for 4 screened scenarios. Sub-scenario frequencies were defined as the sum of the fire frequencies of fire sources grouped within the sub-scenario. Conditional CDFs for the sub-scenarios were estimated in the same manner as that for the screening analysis. Unavailabilities of front line systems were evaluated, reflecting that only safety related equipment associated with the sub-scenarios was damaged. Severity factor was applied to sub-scenarios due to equipment ignited fires, which was determined as functions of the distance between the fire source and the target equipment(4). Geometry factor was applied to sub-scenarios of transient fuel ignited fires which was defined as a function of length of target cable tray, critical radius of transient fuel and area of fire zone. Critical radius were determined by the fire progression analyses with COMPBRN-III e(5). The non suppression factors(6) were determined based on the time margin for fire suppression analyzed by COMPBRM-IIIe considering the fire protection environment in the sub-scenarios. Fig. 2 presents fire frequencies and CDF for the sub-scenarios. Although this study was a limited fire PSA for seven particular areas, CDFs for fire sub-scenarios were estimated relatively small with the maximum of 10^{-8}/ry. This is mainly because fire occurrence frequency is quite small in Japan and the plant studied has safety systems with highly developed physical train separation.

4. CONSIDERATION

Through this study, we could accumulate the experiences of the application of the fire PSA methodology to the Japanese plant and could confirm the applicability of the methodology itself. However, some key parameters such as severity factors and non suppression factors were derived not from Japanese operating experiences but from US data. This chapter gives some consideration on these parameters based on Japanese specific circumstances, referring to our future work.

4.1. Fire frequencies

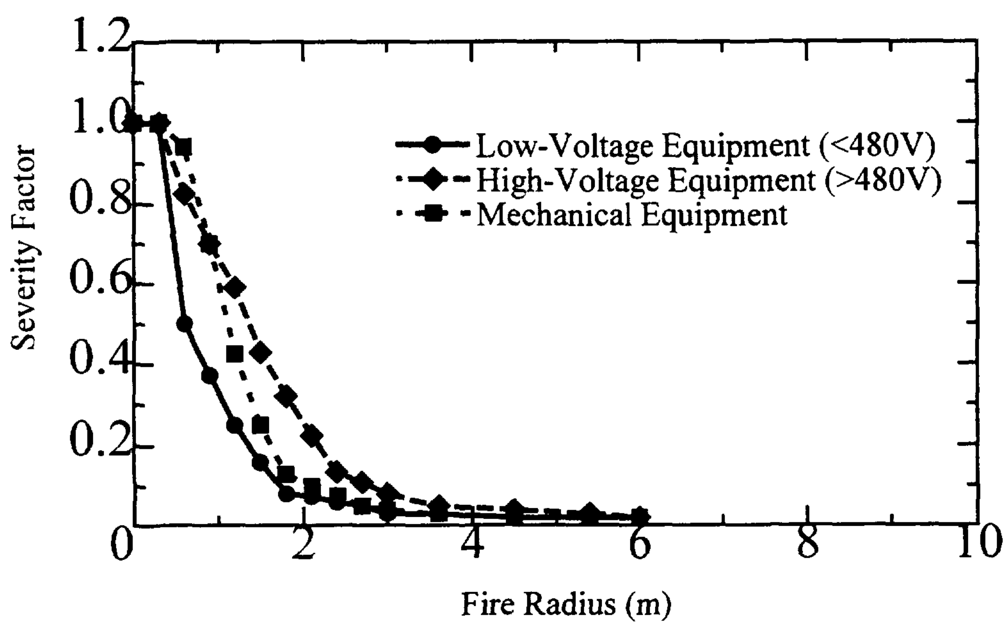
Fire frequency for the scenario was estimated considering the number of components as fire sources in the fire area. In this study Japanese fire experiences until March 1996 were used to estimate Japanese generic fire frequencies based on the Baysian update technique where the US fire experiences until 1992(7) was used as prior distributions. We identified five fire events from our event databases which were applicable to the assessed plant under full operating condition. These were two generator fires, two cable fires and one HVAC fire. Table 1 denotes Japanese generic fire frequency every fire source in comparison with US one. The fire frequencies of fire sources, which had no experience in Japan, were estimated to be $10^{-3}10^{-4}$/ry depending on the prior distribution through the Baysian update. On the other hand Japanese nuclear power plants have been designed and constructed according to the Japanese fire protection code, where flame-resisting or noncombustible materials have been used for cables and other components in the safety system. For example, switch-gears within a reactor building should be oil-less ones, and the quantity of lubricating oil in a pump should be minimized during power operation. So that JEAG4607(8)(Fire protection guideline for nuclear power plant prepared in industries side) assumes, as fire hazard sources, pumps with comparatively large quantity of lubricating oil such as RCP, charging pump and turbine-driven pump, and power cable and so forth. As a result, motor-driven AFW pump, motor-driven valve and control & instrumentation cables are excluded from the fire hazard sources in fire safety evaluation. In order to reflect these Japanese specific circumstances and to get more realistic fire frequencies in fire PSA, some consideration such as more detailed categorization for fire sources should be made in the future.

4.2. Fire induced initiating events

We selected five fire induced initiating events based on those of internal event PSA(9), namely loss of PCS(Power conversion system), other transients, loss of CCWS, loss of offsite power and manual trip. In addition, initiating events induced by mal-function of valves due to hot short in electric

TABLE 1. FIRE FREQUENCIES OF COMPONENT TYPES

Fire source	Prior distribution (US data)			Posterior distribution		
	No. of fire events	Fire freq. (/ry)	EF	No. of fire events	Fire freq. (/ry)	EF
Battery	3	2.6E-3	15	0	6.1E-4	9
Battery charger	3	3.4E-3	30	0	4.4E-4	11
Control room	3	3.4E-3	16	0	6.8E-4	9
Diesel generator	60	5.9E-2	14	0	1.6E-3	5
Generator	13	1.6E-2	21	2	3.1E-3	3
Human error	22	2.4E-2	19	0	1.1E-3	7
HVAC	5	4.8E-3	15	1	1.7E-3	4
Logic cabinet	28	2.9E-2	19	0	1.2E-3	5
MCC	11	1.2E-2	17	0	8.9E-4	7
Motor	5	4.4E-3	16	0	7.2E-4	8
Cable	21	2.3E-2	24	2	3.3E-3	3
Pump	30	2.4E-2	6	0	2.3E-3	4
Switchgear	23	2.3E-2	15	0	1.2E-3	6
Transformer(<4kV)	5	5.6E-3	13	0	9.0E-4	7
Transformer(>4kV)	29	2.8E-2	8	0	1.5E-3	5
Turbine	14	2.0E-2	20	0	9.9E-4	9



Docket Nos. 50-498 and 50-499. dated January 21,1992.

Fig.3. Severity factor.

circuits were taken into account. These were PORV(power operated relief valve) LOCA, IS(interfacing system) LOCA and failure of reclose of MSRV(main steam relief valve). The conditional occurrence frequencies of the above fire induced initiating events were obtained by system analyses using fault trees. In principle conditional fire-induced initiating event occurrence frequencies were given 1.0 except for fire-induced initiating events due to hot short. The probability of 0.1 was allotted for hot short in case of a switchgear fire by engineering judgment. This probability should be confirmed theoretically or experimentally considering its control circuit structure(EWD: Elementary Wiring Diagram).

4.3. Propagation scenario

In this trial application CDFs of fire propagation scenarios were estimated comparatively large (in Fig. 2). This is mainly because of large failure probabilities of fire barrier and coarse fire scenarios. For the fire propagation scenarios, fire propagation probabilities were assessed based on fire severity and fire barrier rating. The failure probability of 2 hour rating fire barrier was assumed to be 0.5 for fire severity of 2 hours, which seems too conservative. Further in this study fire scenarios for fire area were defined without subdividing fire area into fire zones, which seems too coarse for propagation scenarios as well as local scenarios. We are now re-evaluating fire scenarios with fire area being subdivided into fire zones.

4.4. Definition of scenario and sub-scenario

In this application fire scenarios were defined only for safety systems because of lack of plant information. However as the fire protection guideline is not always applied to non-safety class components and cables, we must take into account fire scenarios induced by these non-safety components and cables in the future.

4.5. Severity factor

The severity factors represent the probabilities that equipment ignited fires can affect the target equipment. In this study the severity factors, which are shown in Fig. 3, were derived from US data and determined as functions of the distance between the fire source and the target equipment(4). This figure shows that a mechanical equipment fire affects target components with the probability of 0.3 at 1 m from the fire source regardless of its oil possession. Regarding cables, JEAG 4607 asks cables for safety systems to satisfy the IEEE-383 incombustible standard and the IEEE-384 separation standard, where the distance of more than 90 cm is required between cables. However, Fig. 3 shows that electrical equipment has the probability of 0.7 to affect the targets at 90 cm from the fire source. It suggests that the severity factor seems too conservative, at least for safety-graded cables. In order to reflect the above Japanese circumstances, severity factor should be more sophisticated with appropriate categorization distinguishing safety graded cables from non-safety graded cables based on some cable fire experiments and analyses.

4.6. Geometry factor

Geometry factors are applied to sub-scenarios of transient fuel ignited fires (due to human errors) and are defined as follows;

$$Rg=2rL/S$$

where Rg= geometry factor, L= length of target cable tray, r = critical radius of transient fuel, S = area of fire zone or area

Critical radius r were determined by the fire progression analyses with COMPBRN-III e(5) assuming 10 gallons transient fuel for each sub-scenario. The quantity and the entrance frequency of the transient fuel brought in the fire area were estimated based on US data. However, these are quite influenced by the way of plant operation so that these should be based on realistic plant operation.

4.7. Non suppression factor

Fires may not affect target equipment if they are suppressed until target equipment are damaged. Non suppression factor represents the failure probability of fire suppression before the targets are damaged. The non suppression factors were determined based on the reference (6), where the time margins for fire suppression were analyzed by COMPBRM-IIIe considering the fire protection environment in the sub-scenarios. Non suppression factors were applied to only transient fires in this study. The values of key parameters in these calculations, however, were derived from Surry fire PSA(10) so that the applicability of this model should be confirmed based on some fire experiments and analyses in the future.

5. CONCLUSION

Through this study, we could accumulate the experiences of the application of the fire PSA methodology to the Japanese plant and could confirm the applicability itself. In order to make the fire PSA methodology more conformable to the Japanese plants, we are now applying this methodology to the remaining areas such as a main control room, where fires in control panels and logic cabinet are analyzed, together with improving some inconveniences in the methodology found in this application.

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IAEA-SM-345/35

Invited Paper

XA9847508

THE FRENCH FIRE PROTECTION CONCEPT. VULNERABILITY ANALYSIS

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Abstract

The French fire protection concept is based on a principle of three levels of defence in depth : fire prevention, fire containing and fire controlling. Fire prevention is based on arrangements which prevent the fire from starting or which make difficult for the fire to start. Fire containing is based on design measures so that the fire will have no impact on the safety of the installation. For fire controlling, equipment and personnel are on duty in order to detect, to fight and to gain control over the fire as early as possible.

The French fire protection concept gives priority to fire containing based on passive structural measures. All buildings containing safety equipment are divided into fire compartments (or fire areas) and fire cells (or fire zones). Basically, a compartment houses safety equipment belonging to one division (or train) so that the other division is always available to reach the plant safe shut down or to mitigate an accident.

Because there is a large number of fire compartments and fire cells, deviations from the general principle can be observed. For this reason the RCC-I (Design and Construction Rules applicable for fire protection) requires to implement an assessment of the principle of division. This assessment is called vulnerability analysis.

The vulnerability analysis is usually performed at the end of the project, before erection. It is also possible to perform a vulnerability analysis in an operating nuclear power plant in the scope of a fire safety upgrading programme.

In the vulnerability analysis, the functional failure of all the equipment (except for those protected by a qualified fire barrier, designed or able to withstand the fire consequences) within the fire compartment or cell, where the fire breaks out, is postulated. The potential consequences for the plant safety are analysed.

These consequences are mainly loss of both divisions of safety function by common mode failure. Some of them can cause a core meltdown. In vulnerability analysis, these consequences are classified according to 4 criteria :

- common mode failure concerning redundant components of the same safety function,
- Common mode failure concerning support systems of redundant components of the same safety function,
- Selectivity failure,
- Failure of mitigation systems in case of accidental transient phase caused by the fire.

These potential failures are then assess from the safety point of view, by a functional analysis. If these failures have an impact on the means to reach the safe shut down or to mitigate an accidental phase, the potential common mode failure is confirmed and must be treated individually, mainly by wrapping cables with insulation fibbers.

A vulnerability analysis is being implemented on all the French operating plants.

Computerised cable data files are required to select the common mode failures. Functional and fire risk analyses are performed, on a case by case basis, to justify the common mode failures as they are or to provide modifications for the others. An overview of the first results of the vulnerability analysis is given including a typical description of the modifications proposed to improve safety at optimal cost.

INTRODUCTION

This document recalls the main stages of the fire protection concept applied by EDF and then stresses one major point of the French fire protection doctrine: the vulnerability analysis. The vulnerability analysis mainly involves the identification and treatment of common mode failures and so prevents the fire from destroyed both trains of redundant safety systems necessary to reach the safe shut down or to mitigation accidental situations. The practical aspect of this methodology is then explained. The vulnerability analysis is carried out on new projects but also for upgrading operating plants. An overview of the first results of the studies and examples are given.

1. MAIN OBJECTIVES FOR FIRE PROTECTION

Fire protection aims to fulfil three objectives :

- to ensure the safety of personnel,
- to guarantee the availability of the safety systems which are used to shut down the reactor, to maintain long-term subcriticality, to remove the residual heat and to retain radioactivity,
- to limit damage to equipment which could result in long-term unavailability.

The vulnerability analysis brings a significant contribution to the second objective concerning safety systems.

2. FRENCH FIRE PROTECTION CONCEPT

The French fire protection concept is entirely deterministic. It is based on the following two hypotheses :

- a fire may break out anywhere but only one fire at a time,
- the fire may break out whatever the normal operating status of the plant, under power or during shut down, or in a post-accident situation.

The fire protection concept is based on a three level defence-in-depth concept: fire prevention, fire containing and fire controlling.

FIRE PREVENTION

Fire prevention is based on arrangements which prevent the fire from starting or which make it difficult for the fire to start. In concrete terms, this means choosing uninflammable or hardly inflammable equipment and fluids. Ignition sources have to be controlled.

FIRE CONTAINING

If, as a result of human error or of equipment failure, a fire should break out, design measures are taken so that the fire, whatever the effectiveness of the fire-fighting facilities, will have no impact on the safety of the installation. The subdivision of buildings into fir compartments and fire cells and the fire barriers set up must therefore confine the fire so that the two redundant trains of a safety system cannot be simultaneously endangered by the fire

FIRE CONTROLLING

If a fire breaks out, the fire barriers confine it to the fire compartment concerned, thus preventing any direct impact on plant safety. It is nevertheless necessary in order to limit the risks of spreading of the fire and to preserve the availability of the plant, to gain control over the fire as early as possible with automatic detection and fire-fighting actions.

3. SUBDIVISION OF BUILDINGS

Containing the fire requires a adequate subdivision of the plants into fire compartments and fire cells, a proper qualification of all the items used as fire barriers and a maintenance and periodic test programme which ensures the continued operability of the corresponding components.

To meet this objective, it is necessary to use layout rules based on physical or geographical separation of components. For plants involving two redundant trains A and B, that means that basically a fire compartment or a fire cell contains equipment of train A or equipment of train B but never A and B together.

Subdivision of buildings is the key-point of fire protection programme and no vulnerability analysis can be performed in a plant where the subdivision of buildings into fire compartments and fire cells has not been properly done, rewieved and finally approved.

4. VULNERABILITY ANALYSIS (REQUIREMENT)

VERIFICATION PROCEDURE

Verifications (studies and on-site visits) are performed on each installation as it is finished. Any singularity which does not conform to the design arrangements mentioned above is identified.

This analysis must list the calorific loads in each room, assess the fire duration by fire compartment or cell and assess the efficiency of the physical or geographical separation of redundant trains.

Because there is a large number of fire compartments and fire cells, deviations from the general principle can be observed. For this reason the Design and Construction Rules applicable to fire protection (RCC-I) requires to implement an assessment of the principle of separation of redundant trains. This assessment is called vulnerability analysis. The vulnerability analysis is carried out on new project but also for upgrading operating plants. The vulnerability analysis mainly involves the identification and treatment of common mode failures and so prevents the fire from endangering both trains of redundant safety systems necessary to reach the safe shut down or to mitigate accidental situations.

COMMON MODE FAILURE IDENTIFICATION

The vulnerability analysis is carried out in each fire compartment or fire cell. A potential common mode may be identified when a, b, c or d criteria are fulfilled in the same fire compartment or fire cell :

CRITERION a)

Safety mechanical components or electrical connections belonging to two redundant trains of the same system performing a safety function,

CRITERION b)

Safety mechanical components or electrical connections belonging on the one hand, to the train of a system performing a safety function, and on the other hand, to systems required to operate the same system of the redundant train,

CRITERION c)

Electrical connections which are supplied by redundant electrical switchboards, and the number of which is such that the selectivity of the protection of these switchboards is likely to be jeopardised. The criterion c) relating to the non-selectivity of electrical protection is only taken into account when a fire is able to reach both electrical connections simultaneously (only the electrical connections present in the same room shall be taken into account).

CRITERION d)

Components, the failure of which, in the event of a fire, is likely to result in an accident or additional operating conditions and components required to perform a safety function necessary for mitigating this accidental event.

COMMON MODE FAILURE TREATMENT

If a potential common mode is detected, it will be necessary to install one or more qualified fire protections or to demonstrate the existence of a functional redundancy, non-affected by the fire, able to perform the safety function endangered by the fire.

EXAMPLES OF IDENTIFICATION OF COMMON MODE FAILURE ACCORDING TO CRITERIA a, b, c or d

REDUNDANT COMPONENTS OF THE SAME SAFETY FUNCTION (a)

Example: safety pumps of redundant train located in the same room.

SUPPORT SYSTEMS (b)

Example: cooling systems of safety pumps of redundant train located in the same room.

SELECTIVITY FAILURES (c)

Example: Five medium voltage (380V) cables are running through the fire cell 81 (900MW series). They supply a pump, 3 fans and a heating resistance. Each cable is fed by a circuit breaker. The overall amperage of breakers (380V) is 8950 A. The rate of the breaker of the emergency power supply (6600V) is 8000A. In this case the fire can cause loss of emergency power supply by selectivity failure.

Assessment: emergency power supply feeds two other safety fans which are not located in this fire cell. The cables supplying the redundancies of these 2 fans are not running through this fire cell.

Conclusion : loss of emergency power supply does not involve any new common failure mode.

MITIGATION SYSTEMS IN CASE OF ACCIDENTAL TRANSIENT PHASE INDUCED BY FIRE (d):

Examples :

- spurious activation of a pressuriser relief valve,
- opening of a main steam line relief valve caused by the effect of the fire on cables of pressure sensor.

5. VULNERABILITY ANALYSIS (PRACTICAL METHODOLOGY)

The plant has been properly divided into fire compartments and fire cells. It is assumed that the fire will not propagate from one cell or compartment to another. So this vulnerability analysis is performed inside each compartment or cell and common mode failures are identified inside the cell or the compartment

The vulnerability analysis only deals with equipment and cables necessary to perform safety functions. In practice the vulnerability analysis is performed, at the first stage on all the safety-related equipment and cables.

CABLE DATA FILES

To perform the vulnerability analysis it is necessary to know exactly which pieces of safety equipment are lost if a fire breaks out in a compartment. For mechanical components it is possible to do it manually from drawings and on-site visits, but for fire compartments and fire cells containing hundreds of cables, computerised cable data files are required.

The files report the following information for each cable :

- cable route: fire compartment or fire cell, race, tray,
- system involved, division, voltage, power,
- reference of the equipment at both ends of the cables.

These data files are implemented as follows :

Basically several cable files are available at the Instrumentation and Control Engineering Companies which use them for on-site implementation. These files must be updated with modifications

For old plants, these files do not cover all the safety cables and they must be upgraded by existing drawings. If no drawings exist, the file must be built by on-site visual observation. Some special tools have been developed to follow the cable route through penetrations or in trays involving a high number of cables by a French Company AINF. These tools allow us to carry out the work of detection even during full operation of the plant.

The files must be reviewed and approved as follows :

- it must be verified that all the safety cables are registered on the files,
- a test is made on a random selection of cables. A comparison is made between on-site arrangement and files. The deviations are analysed and classified into several categories according to safety impact. The rate of deviation must be less than a few per cent.

IDENTIFICATION OF POTENTIAL COMMON MODE FAILURES

By processing data, it is easy to list all the equipment and cables which are running in one fire compartment or fire cell.

This data processing is done for all the fire compartments and fire cells. Basically, a compartment or a cell contains a majority of equipment of one train: A for instancc. The operator must then focus on equipment of train B which must be carefully and completely listed.

The potential common mode failures are generated from this list by using criteria a and b with the conservative assumption that all the equipment and cables (except for those protected by a qualified fire barrier, designed or able to withstand the fire consequences) within the fire compartment or fire cell, are lost.

In addition, through this process, the operator is able to know precisely which safety functions are lost, or could be lost in case of fire in a given compartment. This is useful for implementing detailed procedures for operating the plant in case of fire.

COMMON MODE FAILURE PROCESSING AND ASSESSMENT

At the beginning of the use of this methodology a few years ago, the operator picked out the equipment of the train in the minority and systematically protected it by fire-rated wraps. To day there are four steps of study before deciding to protect the common mode failure. The aim of each step is to examine if the common mode failure could be accepted as it is or not, and in that case to identify cost optimised modifications.

STEP 1: POTENTIAL COMMON MODE FAILURE

Potential common modes which are identified by data file processing are called potential common modes.

Example : Fire cell 280 in Nuclear Auxiliary Building :

- 2 fans train A and B in that fire cell,
- cables supplying these 2 redundant fans are running in the fire cell.

STEP 2: CONFIRMED COMMON MODE FAILURE

At this second stage a functional analysis is performed: a potential common mode failure is confirmed if it causes safety consequences. The loos of safety function must be analysed in normal state and post accident situation of the plant.

Example : Fire cell 280 in Nuclear Auxiliary Building

Step 1

- 2 fans train A and B in that fire cell
- cables supplying these 2 redundant fans are running in the fire cell.

Step 2

These fans are blowing fresh air in 2 rooms housing high pressure safety injection pumps. Failure of these 2 fans causes overheating of the pump room, failure of both pumps and unavailability of the high pressure safety injection system.

This common mode failure has safety impact so a fire risk analysis is performed at the next step.

STEP 3: FIRE RISK ANALYSIS

At this step, an assessment of the first assumption: "the functional failure of all the equipment (excepted those protected by a qualified fire barrier, designed or able to withstand the fire consequences) within the fire compartment or cell where the fire breaks out is postulated" is done. The mechanism of fire propagation is analysed considering the possibility of

- Flash over,
- Local fire (cables or equipment),
- Plume or heat radiation,
- Mutual aggression of components.

At this step, it could be decided to implement complementary fire protection measures (sprinklers, screen) to avoid the fire to endanger both trains.

Example : Fire cell 280 in Nuclear Auxiliary Building

Step 1

- 2 fans train A and B in that fire cell,
- cables supplying these 2 redundant fans are running in the fire cell.

Step 2

These fans are blowing fresh air in 2 rooms housing high pressure safety injection pumps. Failure of these 2 fans causes overheating of the pump room, failure of them and unavailability of the high pressure safety injection system.

This common mode failure has safety impact in mitigation of small size pipe failure. A fire risk analysis is performed at the next step.

Step 3

TABLE I. MECHANICAL AND CABLE COMMON MODE FAILURES

Mechanical common mode failure :

Fire propagation		Remark	Modification
Flash over :	Yes	The fire cell includes a room with 2 vertical cable races. This lay out condition can encourage fire propagation and flash over.	These races are encapsulated in a metallic casing to limit the burning rate. The natural air cooling of the cables is possible
Local fire (fans)			
Plume	No	Fans are installed on the floor	
Heat radiation	Yes	The fans can be endangered by heat radiation generated by local combustion of nearby horizontal cable race	These cable races are encapsulated in a metallic casing to limit the burning rate and radiation heat
Mutual aggression of components	No	Fans are separated by a ventilation plenum	

Cable common mode failure :

Fire propagation		Remark	Modification
Flash over :	Yes	The fire cell includes a room with 2 vertical cable races. This lay out condition can encourage fire propagation and flash over.	These cable races are encapsulated in a metallic casing to limit the burning rate. The natural air cooling of the cables is possible
Local fire (cables)			
Plume	No		
Heat radiation	Yes	The cables supplying the fans can be endangered by heat radiation generated by local combustion of nearby horizontal cable race	These cable races are encapsulated in a metallic casing to limit the burning rate and radiation heat
Mutual aggression of components	No	Cables are separated by distance	

For this case there is no fourth step because a solution has been found in step three.

STEP 4: DETAILED FUNCTIONAL ANALYSIS

Step 2 has confirmed that a safety function has been lost. The functional analysis performed at this step is a detailed functional analysis whose aim is to find whether an other safety system can perform the lost safety function.

TREATMENT FOR COMMON MODE FAILURES

The Treatment for common mode failures requires high technology equipment. The most frequently adopted solution is to wrap the cables with insulation fibbers.

EDF' specifications have been written up for component testing. For cable wraps, fire tests in laboratories take into consideration the energy dissipated in the cable by the Joule effect, this energy being provided by an electrical resistance installed in the neighbourhood of the cable inside the wraps. Tests for assessing the long term behaviour and ageing are also required.

Most of plants use a soft wrap of mineral fibbers designed by the French Company Mecatiss. This equipment is easy to install in all the configuration met in operating plants. The rating is available in the range 30 to 180 minutes and is chosen according to the design fire duration of the compartment or cell.

6. OVERVIEW OF VULNERABILITY ANALYSIS IMPLEMENTATION

RULES

For French nuclear power plants under construction (N4 series), the design rules concerning fire protection have been issued by EDF, approved by the Safety Authority and implemented since the beginning of the design.

These rules are formalised in the Design and Construction Rules applicable to fire protection (RCC-I). For N4 series, the vulnerability analysis has been carried out at the design phase.

For operating plants (Fessenheim, Bugey and the series CP1, CP2, P4, P'4), the original fire protection had been designed in accordance with less complete rules.

The first revision of RCC-I, issued in 1983, only mentioned criterion a. Criteria b and c have been introduced in revision 2 issued in 1987. Criterion d was introduced in revision 3 issued in 1992.

An upgrading programme is in progress in all the operating plants to include most of the requirements of the RCC-I issued in 1992. The set of rules concerning fire protection which takes these developments into consideration is formalised in the Fire Directives applicable to each plant or series.

IMPLEMENTATION: FIRST RESULTS

The first results (see appendix for details) show that :

55% of the common mode failures are accepted as they are,

32% are treated with wraps,

5% are treated with different solution,

8% are still in process.

7. CONCLUSION

Protection against the risk of fire is based on requirements involving three levels of defence in depth.

To ensure that the design hypotheses are fulfilled, a verification phase including a vulnerability analysis is performed at the end of the project.

The vulnerability analysis assumes a conservative assumption that all the equipment and cables (excepted those protected by a qualified fire barrier, designed or able to withstand the fire consequences) within the fire compartment are unavailable and identifies the common failure mode.

Computerised cable files are required to perform this task. This methodology requires a significant contribution of functional and fire risk analysises and allows us relevant cost reduction on series.

The vulnerability analysis increases the operator's confidence for the efficiency of the compartmentation.

In case of fire this methodology guarantees the availability of the safety functions and provides information about availability of safety systems to plant operators.

Appendix

COMMON MODE FAILURE (CMF)

FUEL BUILDING

Potential CMF	Confirmed CMF	Acceptable CMF	Treatment	Cable wrapping
22	3	1	2	2

NUCLEAR AUXILIARY BUILDING

Potential CMF	Confirmed CMF	Acceptable CMF	Treatment	Cable wrapping	Metallic wrapping	Cabinet wrapping	Sensor wrapping
29	9	0	9	4	1	3	1

REACTOR BUILDING

Potential CMF	Confirmed CMF	Acceptable CMF	Treatment
56	25	later	

SWITCHGEAR BUILDING

Potential CMF	Confirmed CMF	Acceptable CMF	Treatment	Wraps	Cable rerouting	Screen
235	117	1	116	102	9	5

PILOT FIRE RADIUS SIZE AND ITS VARIATION REGARDING THE UNCERTAINTY IN FIRE RISK ASSESSMENT

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Abstract

The impact of a combustible load with limited amount of heat on the characteristics of fire generated local environment is considered. The combustible load apportionment on the floor and its ability to release the heat at a different rate regarding the temperatures and heat flux in zones formed in the NPP compartments is studied using calculations. Several ways of variation of a pilot fire radius in the same range are compared.

1. Introduction

To evaluate more realistically the damages that might occur in the critical fire scenarios specified for NPP compartments the deterministic physical models are usually used simultaneously with probabilistic methods. A commonly used tool in the field of fire PRA is the computer code COMPBRN the last version of which is described in reference [1]. According to reference [2] about 15 parameters are significant among the numerous ones determining the influence of a fire generated local environment on a vulnerable component. These parameters can be divided in several groups. The first group includes parameters such as dimensions of compartment whose members can be considered as constants for all fire scenarios taking place in a given NPP compartment. The second group includes parameters with large states of knowledge uncertainties, such thermo-physical properties of materials, burning efficiency, damage temperature and so on, that are represented by distributions rather than by point values. The last group includes parameters, such as the radius of the pilot fire and burning rate of combustibles involved in a flame, that are very significant regarding the way a fire grows. The values of these two parameters are closely related to the temperatures of flame, plume and hot gas layer (HGL) as well as the depth of HGL, released heat flux, etc. The temperatures of separate zones of a fire generated environment and the released heat flux strongly determine the damages due to a specific fire scenario.

Reference [3] shows that between the size of the burning pool and the current value of pyrolysis rate a strong (physical) relation exists. From this point of view the impact of the fire scenarios taking place because of a flammable material of a known type and quantity on the size of the pool and on the pyrolysis rate can be easily evaluated. It is not a common case in NPP compartments where many objects with unknown combustible loads and burning rates are located. To assess the impact of the fire scenarios the radius of a pilot fire is usually assumed according to a hypothesis for fire of the size that is expected. Once the total amount of heat expected to be released at combustion and the fire size are specified the degree of expected fire severity can be represented using different types of a flammable material.

In the case of a pool fire the quantity of flammable material involved in combustion will determine the value of its radius. The variation of the pool radius during the fire growth can be evaluated approximately. A possible way is the current quantity of specified flammable material to be 'put' in a volume between a simple 'top surface', for instance part of a sphere, and 'bottom surface' such as the floor. To model the flame size and its likely variation during the combustion of an object with unknown burning rate the constant, increasing and decreasing radius of a pilot fire can be used. In case the bound values for a pilot fire radius corresponding to the specific fire scenario are defined it is important to know how to describe the values of the radius inside the range. The random variation of the radius in a corresponding range, as for parameters of the second group, or variation based on physical relations is possible to be applied.

2. Description of the scope

To trace how the choice of a pilot fire radius and burning rate of combustibles influences the time to damage of a 'normal' cable one specific situation is studied to avoid abstract reasoning. Compartments of medium size, X = 10 m, Y = 5 m and Z = 4 m, are considered. A flammable object with total combustible load only of about 450 MJ and unknown burning rate is assumed to generate this heat in a compartment. The total amount of heat that will be released during combustion is approximated by a corresponding quantity of different types of flammable liquids. According to ref. 3 the net heat of combustion for crude oil is 42.7 MJ/kg and for heptane is 44.6 MJ/kg. From this point of view the combustible load may be represented by equivalent of either 10.1 kg of heptane or 10.5 kg of crude oil. It is very important how the equivalent quantity of flammable liquid will be distributed on the 'bottom surface' because of the dependency between the pool size and pyrolysis rate. The burning rate of the flammable object is defined approximately when using the hypotheses about the expected time for the total heat of combustion to be released. Assuming an expected time for release of the total amount of heat - τ_r s and a radius of pilot fire - R_f m the averaged value of the pyrolysis rate can be evaluated. The averaged mass burning rate m_m kg/s and pyrolysis rate m_s kg/m^{2}s are obtained by simple relations (1) and (2). Equation (3) recommended in ref. [2] is also used.

m_m = amount of equivalent flammable liquid / expected time for release of heat (1)

$m_s = m_m/S$ (2); $S = \pi \cdot R_f^2$; $m_s = m_i(1 - \exp(-2 \cdot R_f \cdot k))$ (3)

The constants $\boldsymbol{k}$ and $\boldsymbol{m_i}$ for the corresponding equivalent flammable liquid are obtained in ref. [3].

Table I shows the distributions and ranges used to describe the state of knowledge uncertainty about parameters of the 'second group'.

3. Choice of pilot fire radius and type of equivalent flammable liquid.

The hypothesis for pilot fire with a constant radius during combustion is considered first.

The hypotheses about the expected time for the total heat of combustion to be released are used to define approximately the radius. It is assumed that the combustible load is released for more than 720 s with mass burning rate equivalent to the pilot fire of a crude oil pool. The constants $\boldsymbol{k}$ = 2.8 1/m and $\boldsymbol{m_i}$ = 0.045 kg/m^{2}s for crude oil are obtained from the ref. [3]. The averaged value of mass burning rate of 10.5/720 can be obtained by means of relation (1). Using relations (2) and (3) the value of pilot fire radius $R_f < 0.35$ m can be estimated. The values of HGL temperature and depth as well as the time to failure of a cable are calculated using an improved version of the programs reported in ref. [4]. The programs use the response surface approximations for these entities presented in ref. [2]. Distributions of any one of the parameters shown in table 1 are described by means of 200 point values generated using Latin hypercube sampling. In the case of crude oil pilot fire with radius R_f = 0.35 m, having a pyrolysis rate $m_s = 0.0387$ kg/m^{2}s, all 200 calculated values for HGL temperature fall in the range of 385-490 K. The above mentioned programs are used to calculate the time to damage of a 'normal' cable. Distributions used to describe the state of knowledge uncertainties about damage temperature and thermo-physical properties of a 'normal' cable are shown in a table I. A cable tray with a depth of 0.076 m, immersed in the HGL and placed at a distance 3 m from the flame is considered. The calculations show that a heat flux on the cable surface for a distance of 3 [m] between the flame and the cable is in the range of 3.6 - 4.0 kW. In this case the damage temperature will be reached only in 4 of 200 trails.

Let us now assume a pilot fire with the same radius that releases the same total amount of heat (450 MJ) but at the burning rate of heptane. For heptane the values of constants k = 1.1 and $m_i = 0.101$ are applied. Using equation (3) the pyrolysis rate of $m_s = 0.0542$ kg/m^{2}s is calculated. The time $\tau_r = 484$ s is obtained using the relations (1) and (2). The calculated values of the HGL temperature and the heat flux on the cable surface fall in the ranges 420-580 K and 5.57-6.54 kW if only the pyrolysis rate is changed in the input data. The number of trials that show the cable temperature is smaller than the damage one are 34 in this case. The number of tirals showing no damage are 52 if the same values for the heat flux on the cable surface, as calculated for oil pilot fire, are used. Taking into account the above it can be seen that if the same pilot fire radius is used the choice of burning rate of the flammable object is significant to the damages of vulnerable components.

In case the oil pilot fire with radius 0.5 m is considered the HGL temperatures fall in the range 580-630 K. If cable - flame distance is 9 m the heat flux on the cable surface is in the range 260-280 W. When the cable is immersed in HGL, no matter the low value of external heat flux, the damage temperature is reached in 190 of 200 trials. The calculations show that in the case of heptane pilot fire with the same radius the HGL temperatures fall in the higher range of 650-735 K because of its higher severity. The trials without damage are only 3 or this pilot fire will lead definitely to failure of cables involved in a hot gas layer. The values of time τ_r for two pilot fires corresponding to heptane and crude oil mass burning rates are 316 s and 190 s. The flammable object, despite its small combustible load, that is able to generate flame radius bigger than 0.5 m or to release the heat faster that 300 s is real a danger for the cables. For flammable objects with limited combustible load the hypothesis of constant pilot fire radius is a very conservative one.

4. Random vs. 'physical' variation of a pilot fire radius

In the above paragraphs an influence of the small combustible load located in a medium size compartment regarding the possibility a 'normal' cable to be damaged was studied. The distribution of the equal combustible load on a 'bottom surface', represented by means of a pilot fire radius, was found to have significant impact on the fire risk. It was shown that if the pilot fire radius is fixed the ability of the flammable object to release faster the heat, represented by the hypothesis of the burning rate, is also an important characteristic. For the compartment, combustible load and 'normal' cable under study the pilot fire radius of 0.5 m was obtained to lead to damage of the cables immersed in HGL. The pilot fire radius of 0.35 m is a critical one for the situation under consideration. The low severity fire, with which heat is released at the lower oil burning rate, cannot damage the cable if it is located at more than 2 m from the flame. Calculations show that with a high severity fire, having the burning rate of heptane, the damage temperature can be reached if the 'normal' cable is immersed in a HGL. In 82-166 of 200 trials the damage temperature is reached when the distance between flame and cable immersed in HGL varies in the range 9-2 m.

TABLE I. CHARACTERISTICS FOR RANDOMLY VARIED PARAMETERS

Randomly varied parameters	Range (oil)	Range (heptane)	Distribution
Heat of combustion - $*E_bH_f$ [MJ/kg]	38.4 - 42.7	38 - 44.6	normal
Compartment ceiling thermal conductivity - k_1 [W/mK]	0.4 - 1.0	0.4 - 1.0	normal
Cable emissivity - ε [-]	0.7 - 0.95	0.7 - 0.95	uniform
Cable thermal conductivity - k_c [W/mK]	0.09 - 0.5	0.09 - 0.5	lognormal
Cable thermal diffusity - α [m^2/s]	$1.8x10^{-8}$ - $7.0x10^{-7}$	$1.8x10^{-8}$ - $7.0x10^{-7}$	uniform
Damage temperature of cable - T_{dam} [K]	450 - 500	450 - 500	uniform

* The symbol E_b denotes the burning efficiency.

TABLE II: RANGES OF THE CALCULATED ENTITIES IMPORTANT TO THE DAMAGE OF A 'NORMAL' CABLE

Ranges of the calculated entities	H1 & H2*	H3 & H4	H5 & H6
Pyrolysis rate - m_s [kg/m^2s]	$3.61x10^{-2}$ - $3.94x10^{-2}$ $5.43x10^{-2}$ - $6.73x10^{-2}$	$3.79x10^{-2}$ $6.07x10^{-2}$	$3.81x10^{-2}$ $6.15x10^{-2}$
Heat flux on the cable surface - +q [W/m^2]	3300 - 4370 5600 - 8500	3730 - 4110 6610 - 7768	3760 - 4410 6730 - 7900
Width of a hot gas layer - L [m]	2.68 - 2.75 2.71 - 2.80	2.716 2.748	2.72 2.75
Temperature of hot gas layer - T_h [K]	464 - 568 525 - 670	490 - 535 555 632	493 - 545 565 640
Temperature on a cable surface - T_c [K]	340 - 500 342 - 500	346 - 500 360 - 500	355 500 379 - 500
Number of trails the damage temperature is reached	116 185	139 192	148 194
Range of a time to damage on condition a cable is HGL - τ_c [s]	38 - 105 24 - 92	53 -80 33 - 105	51 - 91 32 - 112

* The symbols H1 & H2 denote the hypotheses for random variation of pilot fire radius in cases the oil and heptane are burning. The symbols H3 & H4 are used to denote the hypotheses for decreasing radius and by symbols H5 & H6 the hypotheses for initially increasing fire radius are marked.

+ The heat flux on the cable surface is calculated for flame - cable distance equal of 3 [m].

In this study the impact of small combustible loads regarding damages of a 'normal' cable is considered. It is likely the flame size to be changed during the fire growth if limited combustible load is available. The variation of the pilot fire radius in a specified range can be used to represent the uncertainty in expected time the heat of thr combustible load will be released. The time τ_r is practically impossible to be evaluated by a point value. Using an expert opinion this time can be estimated within a range. Assuming the pilot fire radius will be changed during a specific fire scenario the question arises how the hypotheses about the ways of variation influence the time to damage of a 'normal' cable. Two main trends in the flame radius change are likely to occur. A flame radius will decrease during combustion if it involves at once the whole combustible load. A fire will be grown in a different way if the part of the same combustible load covered with flames increases until the moment when the whole combustible load is involved. In such a case the flame radius will increase initially and then will start to decrease.

Hypotheses for the changes a pilot fire radius in a random variation and the two cases of fire growth mentioned above are compared. The bound values of 0.35 and 0.5 for the range of the pilot fire radius are used. The random variation of the fire radius is depicted using the uniform distribution and 200 point values generated by means of Latin hypercube sampling.

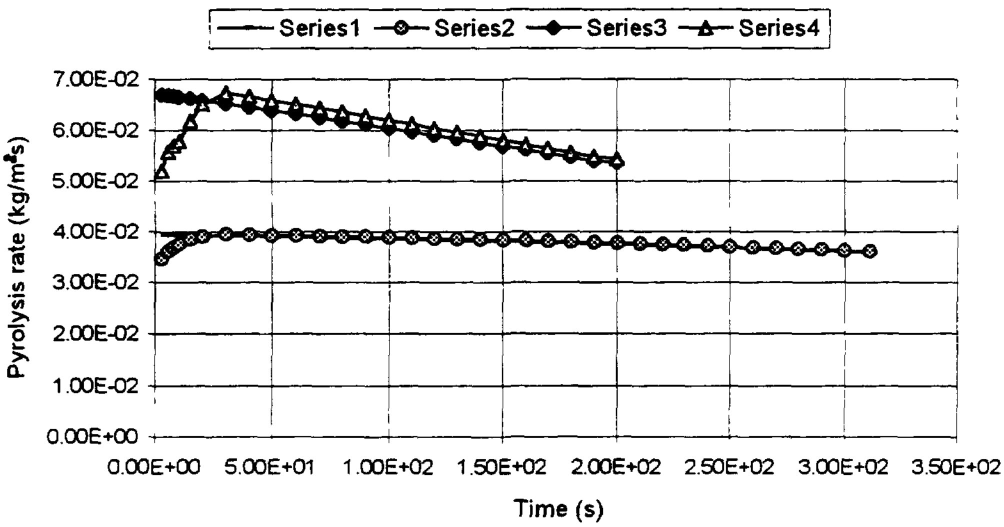

FIG. 1. Variation in the pyrolysis rate versus time per crude oil and heptane (for explanation of symbols, see Table II).

Series 1 - H3; Series 2 - H5; Series 3 - H4; Series 4 - H6

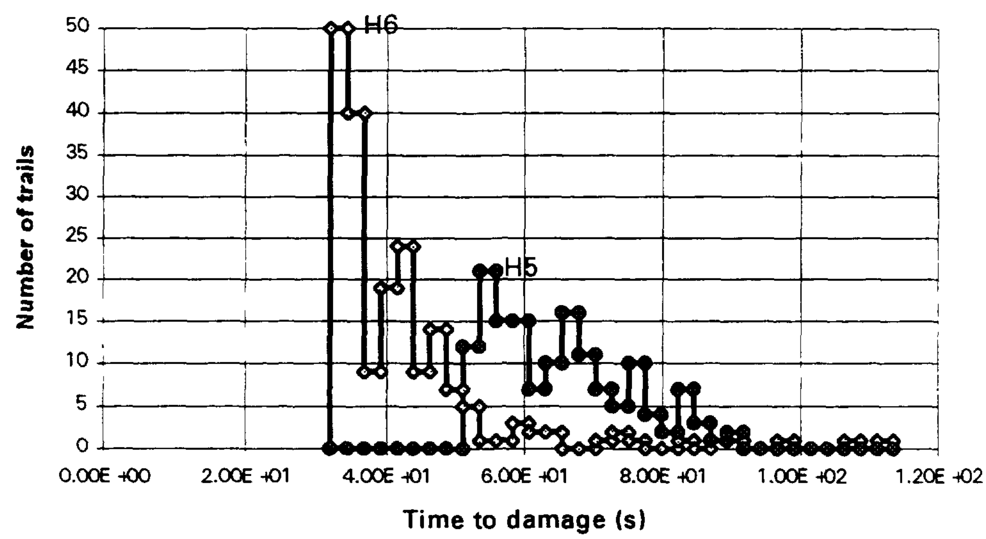

FIG. 2. Distribution of the time taken to damage a cable in the HGL (H5 and H6).

The cases of 'physical' variations of the pilot fire radius in a specified range are evaluated as oil and heptane equivalents of a combustible load 'burning' in a pool. The heptane equivalent of combustible load is used to represent a hypothesis that a time τ_r cannot be outside the range 190 -480 s when a flammable object generates a high severity fire. In the case of decreasing radius the fire is assumed to start at a pool radius of upper bound (0.5 m). The crude oil equivalent is used to study the hypothesis that time τ_r cannot be outside the range 315 -720 s when flammable object lead to a low severity fire.

The inside values of the fire radius range and the corresponding pyrolysis rates are calculated on the basis of the current quantity of flammable liquid equivalent using the programs reported in ref. 4. The values of the fire radius and of the pyrolysis rate are averaged for the time interval the radius reaches the lower bound of 0.35 m. For the case of initially increasing pilot fire it is assumed that the flame starts with 20 % of the flammable liquid and for 30 s involves all the available quantity. Here the values of the fire radius and pyrolysis rate are also integrated over time.

Variations of the pyrolysis rate during the time interval the radii of pilot fires reach the lower bound of 0.35 m for burning of crude oil and heptane equivalents are shown in fig. 1. The flame, plume and HGL temperatures as well as the heat flux on the cable surface are calculated using the corresponding values of the parameters in all 200 trials. For any of the hypotheses about pilot radius variation the parameteres of the 'second group' are presented as the same set of 200 point values generated by Latin hypercube sampling is used. The averaged values of fire radius and pyrolysis rate are applied in all trials that describe the cases of 'physical' variations. The hypothesis for random variation of pilot fire radius in contrast uses a set of 200 point values for fire radius as the corresponding values for pyrolysis rate are calculated by equation (3). The ranges of calculated values for entities important for the damage of a cable are shown in table II. The data of table II shows that the hypotheses for 'physical' variation of the pilot fire radius in the cases of both low and high severity fires lead to damage of a cable in a graet number of trails.

Distribution of the time to damage of a 'normal' cable assuming the pilot fire radius will initially increase (hypotheses H5 & H6) is shown in fig. 2. It can be seen that with both hypotheses the main part of the trails where damage temperature is reached is situated near the lower bound of the time interval.

5. Conclusions

The impact of a limited amount of heat released in a medium size compartment is considered. The distribution of material that releases the heat is reflected by pilot fire radius. The ability of a material to release the heat at different rates is described with a type of flammable material that is chosen. The influence that different hypotheses about size and variation of the pilot fire radius have regarding the possibility a cable to be damaged was traced. Calculations conducted by means of response surface approximations presented in ref. [2] show the damage of a cable is very sensitive to the choice of a hypothesis.

ACKNOWLEDGEMENT

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FIRE SAFETY ANALYSIS: APPLICATIONS

(Session 3)

Chairperson

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A FIRE HAZARD ANALYSIS AT THE IGNALINA NUCLEAR POWER PLANT

XA9847510

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Abstract

The fire hazard analysis (FHA) of the Ignalina Nuclear Power Plant (INPP) Unit no.1 was initiated during 1997 and is estimated to finalise in summer 1998. The reason for starting a FHA was a recommendation in the Safety Analysis Report and its review to prioritise a systematic FHA. Fire protection improvements had earlier been based on engineering assessments, but further improvements required a systematic FHA. It is also required by the regulator for licensing of unit no.1. In preparation of the analysis it was decided to perform a deterministic FHA to fulfil the requirements in the IAEA draft of a Safety Practice "Preparation of Fire Hazard Analyses for Nuclear Power Plants". As a supporting document the United States Department of Energy Reactor Core Protection Evaluation Methodology for Fires at RBMK and VVER Nuclear Power Plants (RCPEM) was agreed to be used. The assistance of the project is performed as a bilateral activity between Sweden and UK. The project management is the responsibility of the INPP. In order to transfer knowledge to the INPP project group, training activities are arranged by the western team. The project will be documented as a safety case.

The project consists of parties from INPP, Sweden, UK and Russia which makes the project very dependent of good communication procedures. The most difficult problems is except from the problems with translation, the problems with different standards and lack of testing protocols of the fire protection installations and problems to set the right level of screening criteria. There is also the new dimension of making it possible to take credit for the fire brigade in the safety case, which can bring the project into difficulties. The most interesting challenges for the project are to set the most sensible safety levels in the screening phase, to handle the huge volume of rooms for survey and screening, to maintain the good exchange of fire- and nuclear safety information between all the parties involved, to assure a good quality assurance during the project and to handle lack of information when making judgements about the reliability of the fire protection devices. The RCPEM is a good help, but the project also demands the involved expertise to be flexible and make good plans to handle the new type of evaluation prob

Long abstract —

1. Introduction

Ignalina Nuclear Power Plant is … one was in commission 1983 and unit two in 1987. Since 1992 … provided aid to INPP in order to increase the safety level. Lot o… to implement the INPP experts in the projects in order to transf… about nuclear safety. The benefit from that is for instance an improvement of the overall safety culture at INPP.

In difference to an average western plant the fire brigade is very comprehensive in order to provide the plant with good fire protection. Instead there are some places where one would expect to find fixed fire fighting systems or barriers in comparison to a western plant.

Therefore it was decided to put in effort to evaluate the fire brigade capabilities in parallel with the main survey.

A large number of fire protection measures have already been performed at Ignalina NPP. For instance exchange of the fire suppression valves, installation of smoke and hydrogen detectors, installation of fire dampers, coating of the turbine hall steel beams and equipment for the fire brigade. There are also ongoing activities such as exchange of a large number of fire doors, exchange of the plastic carpets and additional education and equipment for the fire brigade. However these measures have been decided upon expert judgements and not a systematic fire hazard analysis.

The reason for starting a FHA was a recommendation in the Safety Analysis Report and its review to prioritise a systematic FHA. Fire protection improvements required a systematic FHA. It is also required by the regulator for the licensing of unit no. 1.

2. Organisation (Figure 1)

The project has been built up as a bilateral project between United Kingdom and Sweden. The Project Manager is Mr S. Voronov at INPP. The expert assistance from the western side is organised as shown in figure 2. In order to provide the project with hard data and further assistance the VNIPIET (the Russian design institute) in St. Petersburg is also involved. The organisation of the INPP has been restructured a few times, but the most recent one was doing their initial training course last August. The training course was provided by the western organisation.

As a parallel activity Mr Magnusson has performed checklists, education and screening criteria in order to evaluate the capacity of the fire brigade. This activity was divided in two parts. Part one was to evaluate the capability of the fire brigade at INPP in general and part two was an evaluation of the potential capability to extinguish a fire in specific areas and situations at the INPP.

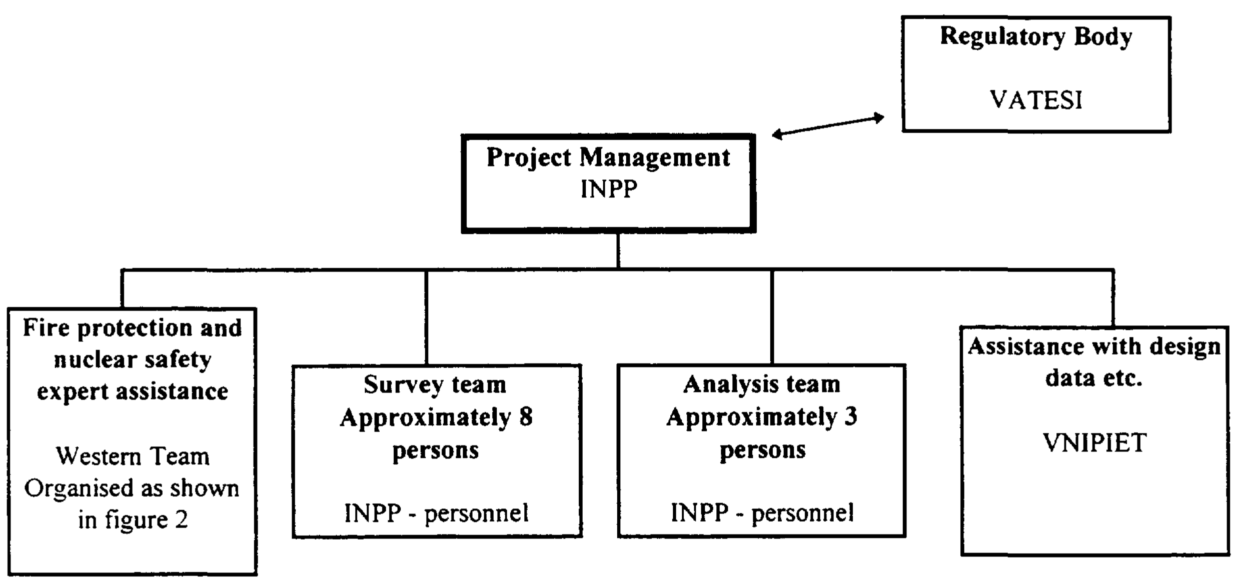

Figure 1. The project organisation in brief.

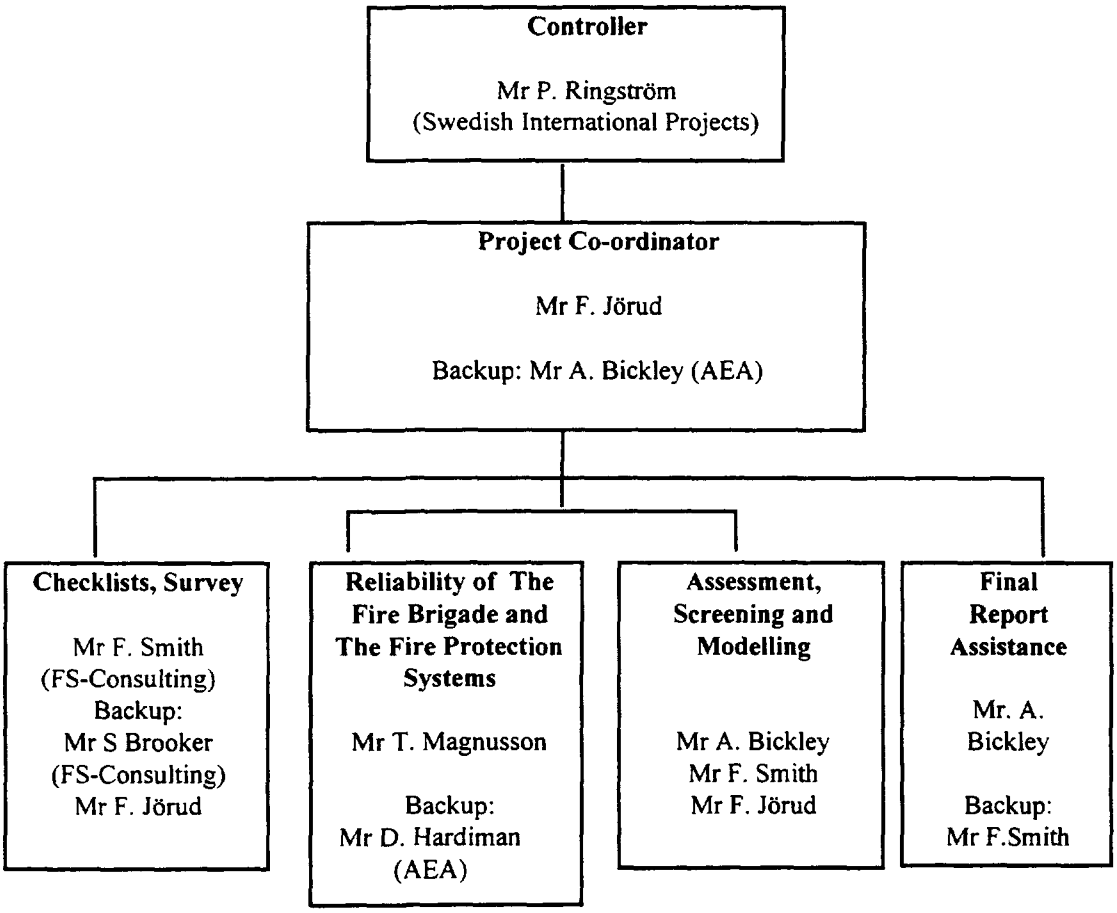

Figure 2. The western assistance team.

3. Evaluation methodology, limitations and time schedule

The requirements from the regulator is to perform a comprehensive fire hazard analysis. Different methodologies have been discussed and IAEA methodology selected, prescribing a full scope deterministic analysis. The IAEA draft of a Safety Practice "Preparation of Fire Hazard Analyses for Nuclear Power Plants" was selected in principle with some limitations related to applicability. However, the IAEA safety practice guide does not determine all details on how to perform the analysis. Therefore the US DOE's Reactor Core Protection Evaluation Methodology for Fires at RBMK and VVER Nuclear Power Plants (RCPEM) is partly used concerning assessment criteria, assessment of manual fire fighting capability etc. Limitations are also that only fire compartments containing safety related equipment were to be involved including the adjacent ones. For this fire hazard analysis it was also decided to enable the possibility to accept manual fire fighting in some cases. A comprehensive PSA has been completed for the plant recently. The system specifications and dependencies and other technical material from that project will be used as input.

The methodology in brief follows the following phases:

- Data collection to prepare the checklists with hard data about the fire protection devices (penetration material, existence of suppression systems etc.) and location of safety related systems.

- Survey, performed as a walk through filling in the checklists for each room involved in the analysis.
- Screening where the areas/rooms are sorted in - needs further consideration/analysis or the area/room fulfils the screening criteria.
- Analysis of the areas required further analysis during the screening phase

The scope of the fire hazard analysis is unit one. The building is enormous and consists of about 700 rooms, but it has been decided to start with 200 rooms. Those rooms are the one containing safety related equipment with the knowledge of today. It is estimated that more rooms will have to be surveyed before the criteria of the project is reached.

The original time schedule for the fire hazard analysis was a completion of the final report before the 1st of March 1998. The estimation of today is to finalise in summer 1998.

4. The main survey team

In order to perform the walk down survey, checklists had to be prepared. There were checklist forms in the RCPEM but they were not found to be suitable on detailed level. The new checklists have the advantage to be partly developed by the survey team and therefore accepted and understood. The training started with the western team explaining the meaning of each item in a checklist template. To exemplify the western team filled in the checklists together with the INPP personnel in a cable spreading room and a battery room. After that the survey teams (three teams led by Mr Voronov, two in each team) had to split up in order to perform a complete survey of one room each. Rooms in vicinity of each other were chosen in order to enable western assistance with translation.

To enable the initial screening the project decided to follow items (1), (2) and (3) in paragraph 202 of the IAEA Guide 50-SG-D2 These items are dealing with ensuring safe shut down, removing residual heat and ensuring that any release are below prescribed limits. Also the single failure criteria was established as described in paragraph 216 of the IAEA Guide. In practice mutually redundant plant performing the same function should (preliminary interpretation, suggested by the western experts) be:

- In different fire compartments segregated by 3 hour barriers or
- In different fire zones separated by more than 6 metres horizontally with no intervening combustibles. Each fire zone should have an automatic fire detection and extinguishing system. or
- In different fire sub-compartments separated by fire barriers with a 1 hour rating. Each fire sub-compartment should have an automatic fire detection and extinguishing system. or
- In the same fire zone but protected by 2 diverse methods of fire detection and fast-acting fire extinguishing system.

These issues are used in the safety principle for existing UK plants and will perhaps be reconsidered during the project.

The screening criteria in the RCPEM were found to be well adjusted when meeting the problems in a Russian designed NPP. The criteria the project found out to be well adjusted were for instance judgements about the doors and penetrations.

A number of different technical documents have been produced or are under production to facilitate the process of screening and modelling. These documents are for instance:

- Screening criteria
- A template for the final FHA report
- Criteria and methods for assessing fire resistance and fire compartment boundary
- A guide on the fire resistance of steel and structural fire resistance
- A guide for determining generic fire loads for electrical cabinets

The main problem so far has been the lack of data from fire tests of the fire protection devices. For instance the cable coating material, dampers, penetration seals etc.

5. Evaluation of the fire brigade capability.

The western team started with an evaluation of the fire brigade using the checklist in the RCPEM. At the same time the team was getting familiar with the fire brigade.

Most of the evaluated subjects were considered as satisfactory. However there were some subjects that need further consideration. Most of the findings that need further evaluation are planned to be upgraded. The main problem in this evaluation is that the fire brigade doesn't have adequate fire fighting equipment available to fight all kinds of fires. For example they don't have high expansion foam equipment to use in areas with high voltage. There are also some problems with personal protective equipment.

The team found the checklist and the screening criteria in the RCPEM -checklist, well adjusted to the problems in Russian designed reactors. As soon as the main survey team notifies an area in need of the fire brigade extinguishing capability, there will be an adjustment of the proposed screening criteria.

6. Evaluation of the potential capability to extinguish a fire

The training of the survey started with the western team trying to explain the meaning of each item in the checklist. To exemplify, the western team filled in the checklists together with the INPP- personnel for a local relay room. The team consisted of plant personnel and personnel from the fire brigade.

The team had to make a special checklist about the capability to extinguish a fire in a certain room. One of the remaining problems is that there are need of judgements to be made of the local fire brigade, but the head of the fire department is at the same time the inspection authority for INPP fire protection compliance. Therefore the fire brigade can only work as input provider in the project. Screening and analysis competence have to be obtained from elsewhere. This is an important role for the Western experts.

The screening criteria that are used for this type of room (for instance a local relay room) which was estimated to be a common type of room in the plant, are as follows:

- Can the plant staff be in the area within 2 minutes?
- Can the plant staff control and contain the fire until the fire brigade arrives (the next 4 minutes) ?
- Can the fire brigade attend and extinguish a fire in < 6 minutes ?

If the screening criteria are not fulfilled the project has to perform further evaluation. The time criteria were based on the fire event to be expected for this type of room. The judgements were made by the involved project expertise.

The main problem here is to provide the analysis with realistic data about the fire brigade capability. The western team has for instance notified an attitude from the fire brigade that nothing unexpected could happen on their way to the fire room, such as nobody meets up and opens the gates to the plant area or somebody has forgotten a tube with welding gas.

7. Challenges

The most interesting challenges for the project are:

- to set the most sensible safety levels in the screening phase,
- to handle the huge volume of rooms for survey and screening,
- to maintain the good exchange of fire- and nuclear safety information between all the parties involved
- to assure a good quality assurance during the project
- to handle lack of information when making judgements about the reliability of the fire protection devices.

The RCPEM is a good help, but the project also demands the involved expertise to be flexible and make good plans to handle the new type of evaluation problems.

IAEA-SM-345/9

АНАЛИЗ ВЛИЯНИЯ ПОЖАРОВ И ИХ ПОСЛЕДСТВИЙ НА БЕЗОПАСНЫЙ ОСТАНОВ ЭНЕРГОБЛОКА С РЕАКТОРОМ ВВЭР-1000

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Abstract-Аннотация

ANALYSIS OF THE EFFECT OF FIRES AND THEIR CONSEQUENCES FOR THE SAFE SHUTDOWN OF A UNIT WITH A WWER-1000 REACTOR.

This paper describes the status of work, and the results obtained, on the problem of safe shutdown of a nuclear power plant with a WWER-1000 reactor (V-320) in the event of a fire. The study focused on Unit 4 of the Balakovo nuclear power plant, which has been in operation since December 1993; it started in 1996 and should be complete by December 1997.

The methodology used for the analysis includes a deterministic approach to the analysis of safe shotdown, incorporating probabilistic safety assessment techniques.

The study is based on the results of a probabilistic safety assessment, an analysis of the spatial distribution of equipment posing a fire risk and of fire-fighting measures at the plant, thermophysical modelling of the development of a large-scale fire, and operational statistics on fire risk factors and fires at plants with WWER-1000 reactors.

АНАЛИЗ ВЛИЯНИЯ ПОЖАРОВ И ИХ ПОСЛЕДСТВИЙ НА БЕЗОПАСНЫЙ ОСТАНОВ ЭНЕРГОБЛОКА С РЕАКТОРОМ ВВЭР-1000.

В настоящем докладе представлено состояние работ и полученные результаты по проблеме останова АЭС с реактором ВВЭР-1000 (проект В-320) при пожаре. Объектом исследования является 4-й энергоблок Балаковской АЭС, эксплуатируемый с декабря 1993 года. Исследование начато в 1996 году, и его завершение планируется в декабре 1997 года.

Методология, использованная для проведения анализа, включает в себя детерминистический подход к анализу безопасного останова, доработанный с учетом методов вероятностного анализа безопасности.

Исследование базируется на результатах вероятностного анализа безопасности, анализе пространственного расположения пожароопасного оборудования и мер противопожарной защиты на АЭС, теплофизическом моделировании процесса развития крупномасштабного пожара, эксплуатационной статистике по факторам пожарной опасности и возгорания на АЭС с ВВЭР-1000.

1. ВВЕДЕНИЕ

В настоящем докладе представлено состояние работ и полученные результаты по проблеме безопасного останова АЭС с реактором ВВЭР-1000 (проект В-320) при пожаре. Объектом исследования является 4-й энергоблок Балаковской АЭС, эксплуатируемый с декабря 1993 года. Исследование начато в 1996 году, и его завершение планируется в декабре 1997 года. Работа проводится специалистами эксплуатационного института ВНИИАЭС, проектного института Атомэнергопроект, института пожарной охраны ВНИИПО МВД России и Балаковской АЭС.

Исследование преследует следующие цели:

- выявление наиболее важных факторов, влияющих на последствия пожара.;
- повышение эффективности средств, обеспечивающих безопасный останов АЭС при пожаре.

Исследование базируется на:

- результатах вероятностного анализа безопасности 4-го энергоблока Балаковской АЭС, выполненного в рамках программы ТАСИС-91,
- детальном исследовании пространственного расположения пожароопасного оборудования, включая трассировку электрических кабелей,
- анализе мер противопожарной защиты на АЭС;
- теплофизическом моделировании процесса развития крупномасштабного пожара в турбинном зале и его воздействия на строительные конструкции и оборудование;

- эксплуатационной статистике по факторам пожарной опасности и возгораниям на АЭС с ВВЭР-1000, эксплуатирующихся в России и на Украине.

2. СОСТОЯНИЕ ВОПРОСА

В настоящее время существует два основных подхода к оценке влияния пожаров на безопасность АЭС.

Первый подход к решению указанной проблемы объединяет различные варианты детерминистического анализа. Они основаны на определении дефицитов безопасности на основе анализа реализации в проекте принципа эшелонированной защиты и их экспертном ранжировании по важности. Такой подход изложен в методике анализа безопасного останова RCPEM [1], разработанной в США. В методике рассматривается определенный заранее, минимально достаточный набор функций систем безопасности, выполнение которых необходимо для приведения блока АЭС в безопасное состояние (функций безопасного останова). Для каждого помещения проводится анализ последствий пожара с точки зрения возможности выполнения этих функций при постулировании отказа всего оборудования, расположенного в рассматриваемом помещении, за исключением пассивных элементов (трубопроводы, теплообменники, обратные клапаны). К категории "уязвимых мест" относятся те помещения, для которых показана невозможность выполнения хотя бы одной функции вследствие потери всех каналов безопасности.

К преимуществам этой методологии следует отнести:

- системный и единообразный подход к решению вопросов пожарной безопасности АЭС, позволяющий находить простой и эффективный, с точки зрения затрат, способ выбора мер, обеспечивающих наибольшее снижение риска;

- несмотря на значительный объем проводимых исследований, четкая организация и планирование последовательности операций всего комплекса работ и мероприятий;

- методология позволяет проанализировать сложную систему взаимодействий и системные логические связи, такие как расположение электрических кабелей и компонентов схемы станции и одновременно выбрать максимально простой аналитический метод устранения уязвимых мест, применяя альтернативные стратегии.

Вместе с тем, консервативные ограничения детерминистического подхода, действующие в "запас" безопасности, значительно увеличивают объем анализа и не всегда достаточно обоснованы. Так, при анализе помещений предполагается, что при возникновении в них пожаров должно реализоваться требование на выполнение всех функций безопасного останова. Однако, в отдельных случаях пожар не приводит к необходимости срочной остановки блока, а в некоторых других случаях несмотря на полную или частичную потерю функций систем безопасности, расхолаживание блока может быть обеспечено работой систем нормальной эксплуатации. Кроме того, в каждом конкретном случае пожара не все функции безопасного останова в равной степени необходимы. Например, выполнение функции по подпитке первого контура требуется только при возникновении аварии с его разгерметизацией.

Другим примером ограничений, приводящих к заведомому "оптимизму" результатов анализа, является исключение из рассмотрения ситуаций, когда при пожаре возникает комбинация нескольких исходных событий, приводящих к крупной аварии или сопровождающихся зависимым от пожара отказом одного или двух каналов безопасности. Такие случаи для АЭС с ВВЭР-1000 формально не относятся к понятию "уязвимость" с точки зрения RCPEM [1], однако, реально могут внести существенный вклад в риск от пожаров.

Второй подход предусматривает выполнение вероятностного анализа безопасности (ВАБ) [2]. Указанный анализ представляет собой комплексную оценку вклада пожаров на АЭС в частоту тяжелого повреждения активной зоны реактора на основе получения оценок частот возникновения пожаров в помещениях основных зданий АЭС и определения их последствий в виде отказов оборудования. ВАБ в силу комплексности подхода является достаточно глубоким и системным инструментом анализа пожарной опасности и оценки последствий пожаров. Он, в принципе, позволяет определить полный перечень факторов (отражающих специфические свойства проекта АЭС), в наибольшей степени влияющих на величину частоты тяжелого повреждения активной зоны, а также произвести их ранжировку по данному критерию. Преимуществом ВАБа является также независимость результатов от какой-либо субъективно принятой шкалы оценок последствий, что, напротив, характерно для детерминистического анализа.

Вместе с тем, выполнение ВАБ для пожаров представляет собой достаточно объемную и большую по затратам и времени задачу, включающую, в частности, разработку вероятностной модели, которая описывает поведение энергоблока при возникновении внутренних исходных событий аварий, являющихся следствием пожаров, а также возможных путей его приведения в безопасное состояние. Другой важной задачей ВАБ является разработка баз данных по частотам пожаров в помещениях, а также по вероятностным характеристикам огнестойкости оборудования и собственно показателям надежности элементов АЭС. Использование при проведении ВАБ исходных данных, не отвечающих объекту анализа, может существенно исказить как абсолютные, так и относительные количественные результаты.

Таким образом, учитывая ограниченность ресурсов на проведение анализа, наиболее адекватная и эффективная оценка влияния пожаров на безопасность АЭС может быть получена путем разумного сочетания обоих упомянутых подходов, т е при принятии допущений, учитывающих с одной стороны специфические свойства проекта, а с другой - особенности моделирования воздействия пожаров на безопасность Такой подход наиболее целесообразен в том случае, когда ранее был выполнен ВАБ первого уровня для внутренних исходных событий исследуемого энергоблока При этом, методология, используемая для проведения анализа, включает в себя детерминистический подход к анализу безопасного останова [1], доработанный с учетом методов вероятностного анализа безопасности [2]

3. МЕТОДОЛОГИЯ

Основные положения и допущения методики состоят в следующем

1 Влияние пожаров в отдельных пожарных зонах на безопасность АЭС характеризуется вкладом пожаров в вероятностный показатель риска - частоту тяжелого повреждения активной зоны реактора, которая определяется для исходного состояния, соответствующего работе блока на мощности

2 Для определения вклада в значение частоты повреждения активной зоны реактора от пожаров в помещениях АЭС в работе применяются вероятностно-логические модели, основу которых составляют деревья событий, деревья отказов и база данных по надежности элементов, разработанные ранее в рамках ВАБ 4-го блока Балаковской АЭС [3]

3 Пожарные зоны определяются как отдельные помещения, совокупности помещений или территории, которые ограничены барьерами с достаточной огнестойкостью или отделены от других ближайших потенциальных очагов пожара пространством, свободным от горючих материалов с расстоянием не менее 6 2 м При этом под достаточной в анализе понимается огнестойкость

ограждений не менее 1.5ч. или огнестойкость не менее 0.75ч. при наличии в помещениях зоны системы автоматического пожаротушения.

4. Отбор пожарных зон для анализа путей безопасного останова проводится на основании совместного рассмотрения частоты возникновения пожара в пожарной зоне, вызываемых пожаром исходных событий и повреждаемого при пожаре оборудования систем, используемых для безопасного останова энергоблока.

5. Значения по частотам пожаров в помещениях АЭС определяются на основе анализа эксплуатационной информации об имевших место факторах пожарной опасности и их прямых последствиях на АЭС с ВВЭР-1000, расположенных на территории России и Украины.

6. Последствия пожаров, с точки зрения их влияния на работу АЭС, определяются в предположении отказа всего оборудования, находящегося в зоне действия пожара, за исключением пассивных тепломеханических элементов (баки, сосуды, трубопроводы, теплообменники, обратные клапаны). Принимается также, что элементы ограждения пожарных зон (стены, перекрытия, проходки, двери) не повреждаются, если они обладают достаточной огнестойкостью. В противном случае постулируется распространение пожара на примыкающие помещения. Исключением из этого правила является моделирование пожаров в наиболее ответственных зонах, например, в районе расположения турбины, где последствия определяются на основании специального расчета тепловых нагрузок.

7. В анализе, дополнительно к потере работоспособности оборудования, непосредственно находящегося в рассматриваемой зоне, а также оборудования, функционально от него зависимого, моделируются отказы, происхождение

которых может быть обусловлено наличием нефункционально связанных цепей. При этом последствия одного из видов подобных отказов (ложной запитки кабелей) анализируются для формирования перечня исходных событий, вызванных пожаром, и для определения возможных путей развития аварии на энергоблоке вследствие пожара.

8. В анализе не рассматриваются сценарии, реализующиеся при возникновении независимых от пожара событий следующего характера:

- отказе системы пожаротушения;
- отказе автоматически срабатывающих элементов ограждения (двери и огнезадерживающие клапаны);
- независимом от пожара нарушении заземления во вторичных цепях трансформаторов тока;
- отказе элементов релейной защиты от коротких замыканий.

Указанные допущения обусловлены малыми вероятностями описанных выше событий и отвечают принципам анализа, изложенным в методике [1]. На последующей фазе работы по анализу безопасного останова после уточнения значений этих вероятностей предполагается снять эти качественные ограничения и перейти к вероятностному принципу отбора моделируемых сценариев пожара.

8. В анализе принят количественный скрининговый критерий, позволяющий исключать из рассмотрения зоны (характеризующиеся частотой пожаров не выше 10^{-7} 1/год) или отдельные сценарии развития пожаров в зонах. К числу моделируемых (т.е. не исключаемых в соответствии с вероятностным критерием) сценариев пожаров в зонах относятся:

▪ сценарии, приводящие к возникновению вторичных исходных событий и характеризующиеся частотой не ниже 10^{-5} 1/год;

▪ сценарии с частотой от 10^{-6} 1/год до 10^{-5} 1/год, при том, что они приводят к исходным событиям и отвечают некоторым дополнительным условиям, касающимся пожарной нагрузки и зон и наличия источников возгораний;

▪ сценарии пожаров в отдельных зонах с частотой от10^{-7} 1/год до 10^{-6} 1/год, при тех же условиях и если в зонах расположено оборудование или кабельные трассы более чем одного канала системы безопасности.

9. Как составная часть выполняемого анализа, для всех отобранных зон и всех моделируемых сценариев пожаров (т.е. тех, которые не были исключены на основании скрининговых критериев из пункта 8) проводится качественный анализ наличия путей безопасного останова в соответствии с принципами, сформулированными в методике RCPEM [1], и дается описание указанных путей. Модели ВАБ применяются для определения путей безопасного останова и их документирования в тех случаях, когда пожар приводит к возникновению исходного события и отказу оборудования, относящегося к более чем одному каналу системы безопасности.

10. Перечень ослабленных мест проекта, т.е. его особенностей, в наибольшей степени определяющих риск от пожаров, составляется на основе совместного рассмотрения результатов, полученных в рамках анализа путей безопасного останова и оценки вклада пожаров в частоту повреждения активной зоны реактора. Результаты анализов значимости и чувствительности, которые выполняются как составная часть оценки частоты повреждения активной зоны реактора, используются для приоритезации найденных “уязвимостей” с точки зрения выигрыша в снижении риска.

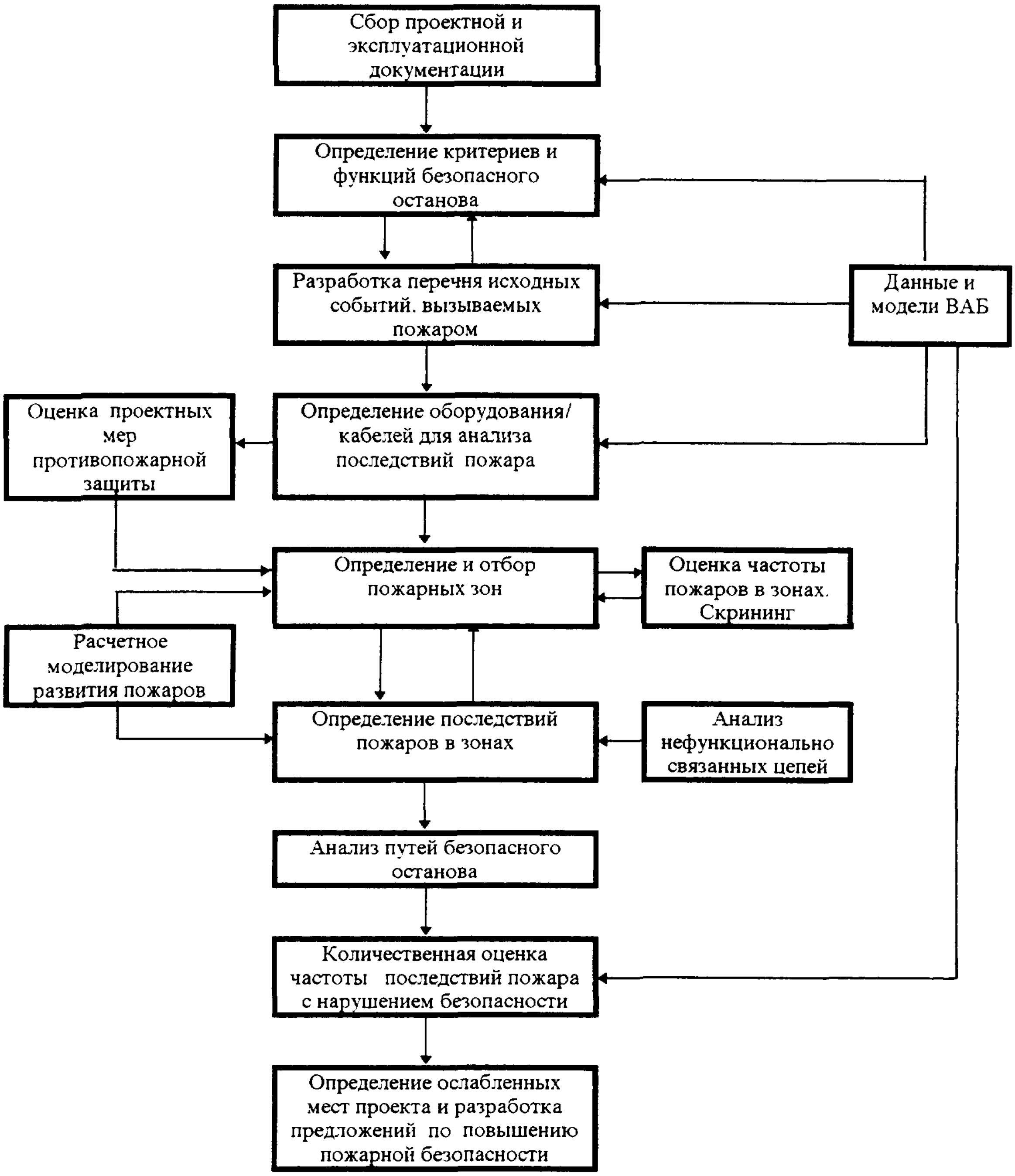

Рис.1. Алгоритм выполнения анализа влияния пожаров на безопасный останов энергоблока с реактором ВВЭР-1000/В-320/

Изложенный выше подход по существу объединяет вероятностный и детерминистический методы анализа, сохраняя преимущества каждого из них При этом следует подчеркнуть, что в отличие от традиционного детерминистического анализа безопасного останова, определение последствий пожаров и поиск путей безопасного останова в соответствии с данным подходом производится на более глубокой основе с привлечением моделей ВАБ. На последующей стадии работы после уточнения недостающей информации, касающейся вероятностных аспектов моделирования сценариев пожаров, возможен переход на полномасштабную вероятностную модель безопасного останова

Алгоритм выполнения анализа влияния пожаров на безопасный останов энергоблока с использованием изложенного выше подхода приведен на рис 1

4. РЕЗУЛЬТАТЫ АНАЛИЗА, ПОЛУЧЕННЫЕ К НАСТОЯЩЕМУ ВРЕМЕНИ

В процессе анализа получены следующие основные результаты

1 Проведен анализ активных и пассивных противопожарных средств и на его основе определены границы пожарных зон Общее число пожарных зон превышает 150 на энергоблок. Наличие источников возгораний, горючих материалов, оборудования систем, выполняющих приведенные выше функции безопасного останова, а также оборудование, отказы которого приводят к исходным событиям аварии являлось основным фактором при отборе помещений блока для их последующей группировки в пожарные зоны (т.е совокупностей помещений, ограниченных пожаростойкими барьерами или свободной территорией). Предполагалось, что локализация пожаров происходит в границах пожарных зон, которые для этого должны обладать достаточной степенью огнестойкости.

Частота пожаров для наиболее важных пожарных зон

Таблица I

Наименование помещения	Число пожарных зон	Частота, 1/год
Машинный зал	1	1.1E-2
Помещения системы аварийного расхолаживания	3	7.9E-4
Помещения маслосистемы главных циркуляционных насосов	2	4.7E-4
Помещения распределительных устройств реакторного отделения	3	4.1E-4
Помещения электродвигателей главных циркуляционных насосов	2	3.2E-4
Блочный щит управления энергоблока	1	2.4E-4
Помещения распределительных устройств турбинного отделения	1	1.8E-4
Помещения панелей управления системами безопасности	3	6.7E-5

2. Проведена оценка частот возгораний от различных источников в каждом помещении, получены частоты возникновения пожара в каждой зоне и на этой основе проведен отбор пожарных зон для более детального анализа.

Статистические данные о распределении возгораний на различных видах оборудования были получены на основе систематизации и анализа эксплуатационной информации за период 1989-1993 гг. для всех энергоблоков с реакторами типа ВВЭР-1000, эксплуатирующихся в России и на Украине. В качестве источников возгорания рассмотрены течи масла, электрическое оборудование и перемещаемые материалы. За указанный период зарегистрировано 596 случаев предшественников пожаров и возгораний. В качестве первичных информационных материалов использованы отчеты о нарушениях на АЭС, годовые отчеты по оценке текущего состояния эксплуатационной безопасности энергоблоков АЭС, карты отказов оборудования АЭС и годовые отчеты о работе АЭС.

Частота возгорания источника конкретного типа в отдельном помещении определялась исходя из его относительной представительности в этом помещении. Частота пожара в отдельных зонах определялась на основании суммарного вклада от различных типов источников возгорания. Частота пожаров для наиболее важных зон приведена в таблице I.

3. Разработан перечень исходных событий, возникающих вследствие пожара, и проведена группировка исходных событий. Ниже

перечислены группы исходных событий, которые могут возникнуть по отдельности или в сочетаниях вследствие пожара:

- останов реактора;
- нарушения нормального отвода тепла через 2-й контур в различных конфигурациях;
- течь из первого контура через уплотнение главных циркуляционных насосов;
- межсистемная течь из первого контура;
- отключение главных циркуляционных насосов;
- открытие предохранительных клапанов компенсатора давления;
- открытие паросбросных клапанов БРУ-А или предохранительных клапанов парогенераторов;
- обесточивание энергоблока (потеря источников третьей категории собственных нужд).

4. Разработан перечень функций безопасности, выполнение которых необходимо при различных исходных событиях, вызванных пожаром, для приведения энергоблока в безопасное состояние. При этом под безопасным состоянием блока понималось состояние горячего останова, при котором отвод остаточных тепловыделений может стабильно поддерживаться в течение 24 часов предусмотренными в проекте системами без экстренных действий оперативного персонала по переключению оборудования, а также состояние расхоложенного блока. Для преодоления последствий указанных выше исходных событий

аварии рассматривались функции безопасного останова, перечисленные ниже:

- приведение и поддержание реактора в подкритичном состоянии;
- поддержание запаса теплоносителя в активной зоне реактора при высоком, среднем и низком давлениях в 1-м контуре;
- отвод тепла через 1-й контур при среднем и низком давлениях, длительный отвод тепла и расхолаживание блока;
- обеспечение герметичности защитной оболочки с целью предотвращения потери среды 1-го контура;
- отвод остаточных тепловыделений и расхолаживание через 2-й контур;
- ограничение роста давления во 2-м контуре;
- обеспечение плотности главного парового коллектора и паропроводов острого пара, изоляция парогенераторов по питательной воде.

5. Разработан перечень систем и компонентов, выполняющих функции безопасного останова при пожаре, и проведен анализ их отказов. Указанные функции могут выполняться как штатными системами так и системами безопасности. Дополнительно в число функций безопасного останова могут быть включены действия, выполняющиеся в качестве мер по управлению запроектными авариями в случае отказа предусмотренных в проекте нормальных и аварийных систем (например, подпитка реактора в режиме "feed and bleed"). Перечень систем и оборудования, отвечающий приведенным выше функциям безопасного останова был составлен на основе моделей ВАБ.

6. Проведен анализ ассоциированных цепей для выявления неявных зависимостей, которые могут возникать при пожаре, а именно:

- вторичных возгораний трансформаторов тока,
- неселективного функционирования выключателей при коротких замыканиях или замыканиях на промежуточную нагрузку,
- распространения пожара по кабельным трассам через неогнестойкие проходки.

7. Проведено теплофизическое моделирование развития пожара и его воздействия на строительные конструкции и оборудование для ответственных помещений. Такие расчеты были выполнены для машзала (пожар на турбине, сопровождающийся горением масла) и кабельного коридора, примыкающего к помещениям всех трех каналов системы безопасности. Результаты расчетов позволили подтвердить приемлемость допущений, принятых в анализе для определения границ указанных пожарных зон.

8. Проведен инженерный анализ последствий пожаров, в том числе из-за коротких замыканий, обрывов и горячих замыканий силовых и контрольных кабелей, приводящих к отключению или ложному срабатыванию оборудования. Были проанализированы все возможные последствия пожаров в пожарных зонах в терминах зависимых исходных событий аварий и отказов элементов систем безопасного останова (учитывались ложные срабатывания и отказы на требование). При выборе моделируемых сценариев развития пожаров в зонах применялся консервативный подход, который обеспечивает рассмотрение наиболее тяжелого протекания аварии. В тех случаях, когда выбор наиболее

тяжелого сценария из нескольких возможных представлялся затруднительным, рассматривались альтернативные сценарии, каждому из которых отвечала определенная условная вероятность реализации.

К концу 1997 года планируется завершить разработку вероятностной модели энергоблока и оценку вклада от пожаров в отдельных помещениях в частоту повреждения активной зоны и окончательно определить "слабые" места в проекте АЭС с ВВЭР-1000/320 с точки зрения обеспечения безопасности энергоблока при пожаре.

5. ВЫВОДЫ

Анализ последствий пожаров на 4-м блоке Балаковской АЭС позволил сделать следующие выводы:

1. АЭС с реактором ВВЭР-1000/В-320 обладает достаточно высокой степенью противопожарной защиты, что объясняется последовательным применением в проекте известных принципов безопасности (защита в глубину, сочетание пассивных и активных средств, канальное построение систем безопасности, физическое и электрическое разделение компонентов, относящихся к разным каналам). Для рассматриваемого проекта невозможны ситуации, приводящие к ядерно опасным режимам.

Пожары в ряде ответственных помещений (здание резервной дизельной электростанции, блочного щита управления и др.) не приводят к значимым с точки зрения безопасности последствиям. Так, пожар в здании резервной дизельной электростанции, в силу принятых компоновочных решений (расположение дизель-генераторов одного блока в разных зданиях), не приводит к возникновению исходных событий и не может вызвать отказ всех каналов систем безопасности. Пожар на блочном щите управления может привести к появлению ложных импульсных (т.е. не обладающих памятью) сигналов, вызывающих изменение состояний оборудования (срабатывание предохранительной арматуры, остановку насосов, изменение состояния запорно-регулирующей арматуры), однако, даже в случае возникновения множественных коротких замыканий на блочном щите управления пожар не сможет повлиять на работу автоматики энергоблока, которая обеспечит переключение оборудования в безопасное состояние вследствие достижения значениями параметров соответствующих уставок. Вместе с тем, оператор сохраняет возможность выполнения всех необходимых функций безопасного останова блока с резервного щита управления энергоблока, который электрически независим от блочного щита управления.

2. В результате анализа были определены потенциально уязвимые, с точки зрения последствий пожаров, зоны, дающие наибольший вклад в величину риска. К числу указанных помещений относятся:

- помещения управляющих устройств УКТС;
- ряд кабельных шахт и полуэтажей;
- помещения датчиков КИП;
- зоны машинного зала (турбоустановка и участок размещения турбопитательных и вспомогательных питательных насосов);
- помещения несистемных комплектных распределительных устройств и агрегата бесперебойного питания.

Типовым последствием пожаров в системных помещениях реакторного отделения является остановка блока по причине отключения главных циркуляционных насосов с отказом одного канала всех систем безопасности. В ряде зон указанное исходное событие осложняется отказом системы продувки-подпитки, обеспечивающей подачу запирающей воды на уплотнения главных циркуляционных насосов. В этих случаях авария при незакрытии оператором дистанционно одной из задвижек на сливе запирающей воды переходит в группу аварий с малой или средней течью из первого контура внутрь защитной оболочки. Для приведения блока в безопасное состояние в этом случае может быть использовано работоспособное оборудование в неповрежденных каналах систем безопасности .

Наиболее критическим из числа системных помещений с точки зрения последствий пожаров является помещение управляющих устройств УКТС 2-го канала систем безопасности. Пожар в данной зоне может вызвать закрытие быстродействующих запорно-отсечных клапанов на всех четырех парогенераторах (по причине ложного формирования сигнала Δt_S>75), что аналогично нарушению нормального отвода тепла через 2-й контур с потерей технологического конденсатора. Указанное исходное событие сопровождается остановкой главных циркуляционных насосов, потерей возможности подачи питательной воды во 2-й и 3-й парогенераторы, а также отказом всего остального оборудования 2-го канала систем безопасности. Предполагается, что наиболее тяжелый сценарий протекания аварии имеет место в случае потери питания и управления сбросных клапанов БРУ-А на 2-м и 3-м парогенераторах после их открытия по повышению давления в парогенераторах, что приводит к фиксации сбросных клапанов БРУ-А в открытом положении.

Опасность такой аварии состоит как в возможности захолаживания 1-го контура в начальный период ее протекания, так и значительного сокращения располагаемых средств по расхолаживанию реакторной установки через 2-й контур Указанная авария в части непосадки двух сбросных клапанов БРУ-А требует специального исследования для обоснования наиболее эффективного способа приведения блока в безопасное состояние и последующей разработки корректирующих мероприятий, например изменения схемы управления БРУ-А Аварии, вызванные пожарами в помещениях управляющих устройств УКТС 1-го и 3-го каналов систем безопасности, протекают аналогичным образом, однако, в отличие от случая, рассмотренного выше, возможна непосадка только одного сбросного клапана БРУ-А

Последствием пожара в машзале (зона турбины) является, как было отмечено выше, нарушение нормального отвода тепла через конденсатор турбины Вместе с тем анализ показал, что для приведения в действие технологического конденсатора в режиме отвода остаточного тепла, необходимо открытие арматуры, дистанционно управляемой с местного щита турбины, что, учитывая ситуацию повышенного стресса, вызванного крупным пожаром на турбоустановке, представляется трудноосуществимым В этом предположении последствием пожара для обеих зон машзала может явиться полная потеря систем второго контура Следует отметить, что относительный вклад указанной аварии в риск значителен из-за большой частоты возникновения пожаров в машзале что требует дублирования управления этой арматурой с БЩУ

Возможным последствием пожара в помещениях системного комплектного распределительного устройства является обесточивание блока, сопровождающееся остановкой главных циркуляционных насосов и штатных систем нормальной эксплуатации Однако, риск подобной аварии невелик, в связи с малой пожарной нагрузкой этих помещений, что не требует дополнительных мер по безопасности

Следует отметить, что во всех указанных выше случаях возникновения пожаров в помещениях АЭС за пределами реакторного отделения, системы безопасности, запитанные от дизель-генераторов, не могут прямо или косвенно быть повреждены в результате их действия.

3. На основе выводов, сделанных по предварительным итогам проделанной работы, наиболее уязвимым местом проекта 4-го блока Балаковской АЭС в пожарном отношении являются помещения управляющих устройств УКТС. Следует также отметить, что снятие ограничений, налагаемых общепринятым в настоящее время принципом единичного отказа для ложной наводки напряжения в нормально обесточенных кабелях, может привести к расширению спектра возможных исходных событий и последствий, в частности, открытию арматуры на линии газоудаления из компенсатора давления, а также более серьезной течи из первого контура через трубопровод системы планового расхолаживания (в последнем случае необходима ложная запитка всех трех фаз силового кабеля в определенном порядке). Вместе с тем, учитывая принимаемую в настоящее время оценку условной вероятности такого типа событий (порядка $5*10^{-2}$ для одной жилы), следует ожидать, что подобные гипотетические аварии не внесут значимый вклад в величину пожарного риска.

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СРАВНИТЕЛЬНЫЙ АНАЛИЗ ОПАСНОСТЕЙ ОБЫЧНОГО И НАТРИЕВОГО ПОЖАРА НА АЭС С БЫСТРЫМ РЕАКТОРОМ

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Abstract-Аннотация

COMPARATIVE ANALYSIS OF THE RISK OF ORDINARY AND SODIUM FIRES AT NUCLEAR POWER PLANTS WITH FAST REACTORS.

A specific feature of a fast reactor is the presence of sodium coolant. Liquid sodium is a combustible material and therefore if any leaks occur in the sodium loops, the escaping sodium may ignite, releasing energy and smoke.

Sodium leaks and fires which have occurred in practice have not posed a threat to the nuclear or radiation safety of reactors. On the other hand, during the operation of thermal reactors, a number of serious fires have occurred, which caused extensive material damage and, if things had turned out badly, could have threatened reactor safety.

In this paper, the authors present a comparative analysis of the risk of ordinary and sodium fires in nuclear power plants. The characteristics of sodium and other combustible materials used in nuclear power plants are analysed [1]. The risk of different types of fires is considered, taking into account features of specific reactor projects, including the location in the reactor building of fire protection systems and combustible material, the fire resistance of fire barriers (screening), and the operation of fire detection and extinguishing systems [2]. Methods of safety probability analysis are employed. It is concluded that the use of sodium at nuclear power plants does not lead to an essential reduction in overall safety.

СРАВНИТЕЛЬНЫЙ АНАЛИЗ ОПАСНОСТЕЙ ОБЫЧНОГО И НАТРИЕВОГО ПОЖАРА НА АЭС С БЫСТРЫМ РЕАКТОРОМ.

Специфической особенностью быстрого реактора является наличие натриевого теплоносителя. Жидкий натрий является горючим веществом, поэтому при разуплотнениях натриевых контуров возможно загорание вытекающего натрия с выделением энергии и дыма.

Случившиеся на практике течи и пожары натрия не создавали угрозы для ядерной или радиационной безопасности реакторов. С другой стороны, при эксплуатации тепловых реакторов произошел ряд крупных пожаров, которые причинили огромный материальный ущерб, а при неблагоприятном развитии событий могли угрожать безопасности реактора.

В настоящем докладе проводится сравнительный анализ опасности обычных и натриевых пожаров для АЭС. Анализируются свойства натрия, а также других материалов, применяемых на АЭС, как горючих веществ [1]. Опасность различного вида пожаров рассматривается с учетом условий конкретных проектов реакторов, в том числе расположения в здании реактора систем безопасности и горючих материалов, пожаростойкости противо-пожарных преград (перегородок), работы систем обнаружения пожара и пожаротушения [2]. Применяются методы вероятностного анализа безопас-ности. Делается вывод о том, что применение натрия на АЭС не приводит к кардинальному ухудшению ее общей безопасности.

1. ВВЕДЕНИЕ

Специфической особенностью быстрого реактора является наличие натриевого теплоносителя. Жидкий натрий является горючим веществом. Температура натрия, находящегося в теплоотводящих контурах быстрого реактора в эксплуатационных режимах превышает его температуру самовоспламенения в воздухе. Поэтому при разуплотнениях натриевых контуров возможно загорание вытекающего натрия с выделением тепла и дыма.

Способность натрия при рабочих температурах возгораться при контакте с воздухом некоторыми специалистами представляется как серьезнейший недостаток этого типа реакторов. На этом основании иногда даже формулируется утверждение о невозможности широкого развития АЭС с быстрыми натриевыми реакторами.

Однако отечественный и зарубежный опыт эксплуатации быстрых натриевых реакторов (около 250 реакторолет) не подтверждает этого пессимистического вывода. Случившиеся на практике течи и пожары натрия не создавали угрозы для ядерной или радиационной безопасности установок. Результаты теоретических и экспериментальных исследовательских работ говорят о малой вероятности крупных течей и пожаров натрия на современных быстрых реакторах.

С другой стороны , при эксплуатации тепловых реакторов произошел ряд крупных пожаров (например, на АЭС Браунз-Ферри в США в 1975 году) , которые причинили огромный материальный ущерб, а при неблагоприятном развитии событий могли угрожать безопасности реактора [1].

Отсюда логически возникает вопрос: насколько опасен натриевый пожар для АЭС по сравнению с пожарами других видов горючих веществ? Рассмотрению этого вопроса посвящена настоящая работа.

2. ОПАСНОСТИ ОБЫЧНЫХ И НАТРИЕВЫХ ПОЖАРОВ

Атомная электростанция является ядерноопасным объектом. Поэтому любые опасные события на АЭС, в том числе пожары следует рассматривать прежде всего с точки зрения их угрозы для ядерной безопасности.

Кроме того, пожар на АЭС, так же как и на любом другом объекте, представляет собой угрозу для людей, материальных ценностей, может повлечь за собой потери от простоя. Следует отметить, что в условиях атомной станции пожар, даже не угрожающий активной зоне реактора, может представить радиационную опасность, если в зоне горения находятся радиоактивные вещества (например, при пожарах в хранилищах сухих отходов, системах спецвентиляции, горении радиоактивного натрия).

Повреждение активной зоны реактора вследствие пожара может произойти при реализации следующих событий:

- разуплотнение первого контура, истечение теплоносителя и обнажение активной зоны, что может быть следствием разрушения вследствие пожара строительных конструкций, опор и подвесок трубопроводов и оборудования первого контура;
- повреждение систем, обеспечивающих контроль и управление протеканием цепной реакции;
- выход из строя систем аварийного расхолаживания

Причинами этих событий могут быть опасные факторы пожара, которые воздействуют либо непосредственно на сооружения или системы, либо на обслуживающий персонал. Опасные факторы пожара являются следствием энерговыделения при горении, выделения дыма и вредных газов, сжигания кислорода.

При обычных пожарах выделяется огромное количество различных вредных веществ. Так, у пенополиуретанов идентифицировано примерно 50 токсичных продуктов горения, у поливинилхлорида около 75, причем некоторые из них обладают канцерогенными свойствами. При горении полимерных материалов (изоляция проводов, материал плат в щитах управления) выделяется хлористый водород, оксид и диоксид углерода, цианистый водород, синильная кислота и другие токсичные продукты горения [1].

При горении натрия образуются его окислы Na_2O и Na_2O_2. Некоторая часть продуктов горения выделяется в виде аэрозолей. Взаимодействуя с влагой воздуха, они довольно быстро превращаются в гидроокись NaOH. Это наиболее опасное из соединений натрия. Предельно допустимая концентрация (ПДК) NaOH в России в воздухе рабочей зоны равна 0,5 мг/м^3. В дальнейшем реагируя с углекислым газом гидроокись постепенно переходит в карбонат Na_2CO_3 , для которого ПДК$_{р з}$ (в воздухе рабочей зоны) равна 2 мг/м^3. Распространяясь в атмосфере, натриевые аэрозоли оказывают воздействие на людей, находящихся на прилегающей территории. Для этого случая ПДК таковы: ПДК$_{м р}$ (максимальная разовая - в течение 30 минут) равна 0,5 мг/м^3 , а среднесуточная ПДКс.с. = 0,05 мг/м^3 [2].

Воздействию натриевых пожаров на окружающую среду уделяется большое внимание. Создан ряд вычислительных программ, позволяющих оценить эффекты такого воздействия. С помощью этих программ могут быть, в частности, установлены для каждого натриевого реактора предельно-допустимые выбросы (ПДВ) натриевых аэрозолей, соответствующие приведенным выше ПДК.

Обычные пожары загрязняют окружающую среду продуктами горения и пиролиза, несгоревшими горючими веществами, которые оказывают негативное влияние на здоровье человека и экосистемы. Опыт обычных пожаров говорит о том, что в 80 ÷ 90 % случаев гибель людей при пожарах происходит от отравления продуктами горения [3]. Экологические последствия пожаров не ограничиваются смертельными исходами. Существует опасность возникновения различных заболеваний, вызванных выделением при пожарах различных токсических веществ.

Анализ показывает, что при обычных пожарах по многим токсическим веществам ПДК превышаются на много порядков. Кроме того, в литературе имеются указания на то, что не все токсиканты, образующиеся при пожарах выявлены, а среди невыявленных могут оказаться обладающие еще более сильным негативным эффектом [3]. Таким образом, выброс в окружающую среду при обычных пожарах на АЭС токсичных и вредных продуктов горения представляют экологическую опасность и должен серьезно анализироваться.

Выход токсичных продуктов при пожаре в аэрозольной или газовой форме также зависит от количества сгоревшего вещества. Кроме того, каждое горючее вещество имеет свою долю выхода. Так, при горении натрия в аэрозоли переходит от 10 до 30 % продуктов горения. При горении таких материалов как нефть и нефтепродукты почти все сгоревшее вещество переходит в газоаэрозольную фазу. Считается, что массовая скорость дымообразования равна массовой скорости выгорания материала [4].

Аэрозоли натрия даже в малых количествах вызывают раздражение слизистой оболочки. Это обстоятельство имеет и свою положительную сторону, т.к. позволяет обнаруживать малые протечки натрия обслуживающим персоналом. Известны случаи, когда операторы обнаруживали натриевый пожар находясь на значительном отдалении от аварийного помещения.

ТАБЛИЦА 1. ТЕПЛОВЫЕ ЭФФЕКТЫ И СКОРОСТИ ГОРЕНИЯ НЕКОТОРЫХ МАТЕРИАЛОВ

Горючий материал или вещество	Скорость горения, кг/м^2·ч	Тепловой эффект, кДж/кг
Бензин	160 - 200	41870
Диз. топливо	150	41870
Древесина	50	13800
Мазут	126	38700
Резина	40	33500
Натрий	30 - 50	10900

Сравнивая опасности обычного и натриевого дыма надо учитывать, что натриевые системы на атомных станциях расположены, как правило, в защитных боксах, а прочие горючие вещества имеются во всех рабочих помещениях.

При пожарах возникает угроза для строительных конструкций, обусловленная тепловыми эффектами горения. В таблице 1 приведены тепловые эффекты и удельные скорости горения некоторых веществ, в том числе натрия. Из таблицы видно, что по этим параметрам натрий занимает одно из последних мест.

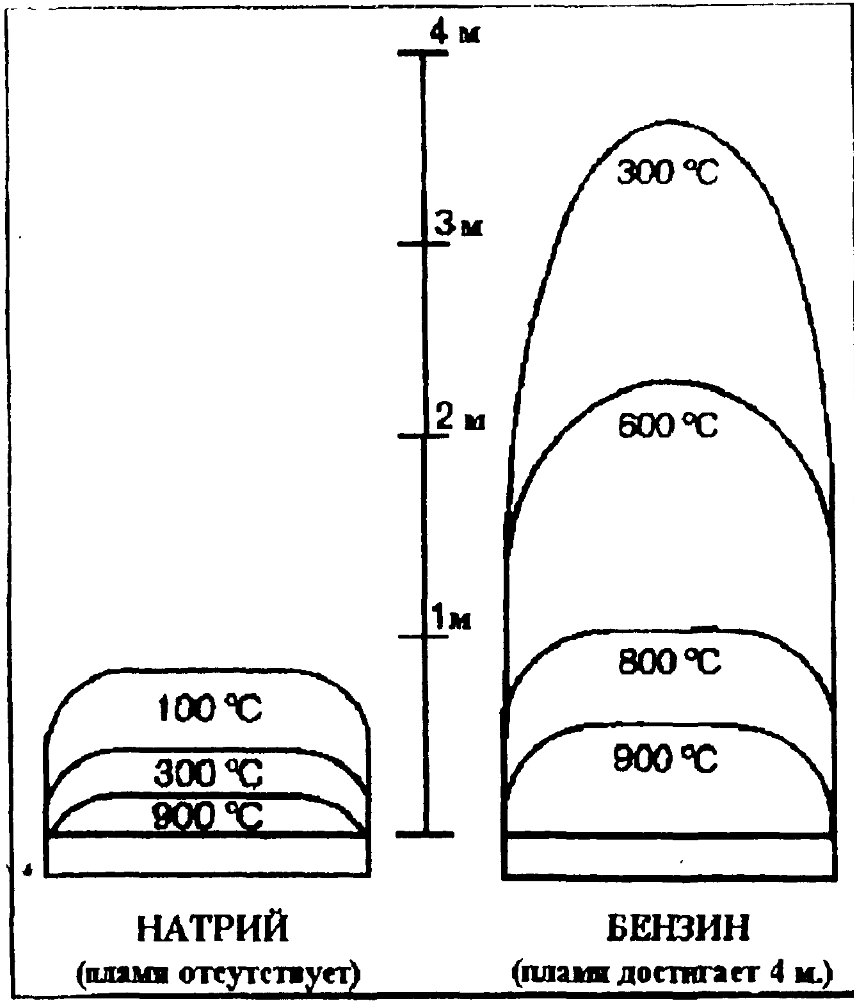
При горении натрия почти не образуется пламени. На рис.1 приведено сравнение температурных полей над горящими натрием и бензином и некоторых характеристик их горения [5].

Температурные эффекты при горении натрия вследствие этого существенно ниже, чем при горении других материалов, применяемых на атомных станциях. Например, в работе [6] описаны полномасштабные эксперименты по изучению пожаров кабелей с изоляцией из поливилхлорида. В этих экспериментах температура газа в помещении превысила 1000°С.

Многочисленные теоретические и экспериментальные исследования показали, что даже при крупных пожарах натрия температура газа в аварийном помещении не превышает 300°С и только в некоторых маловероятных ситуациях поднимается выше этого значения.

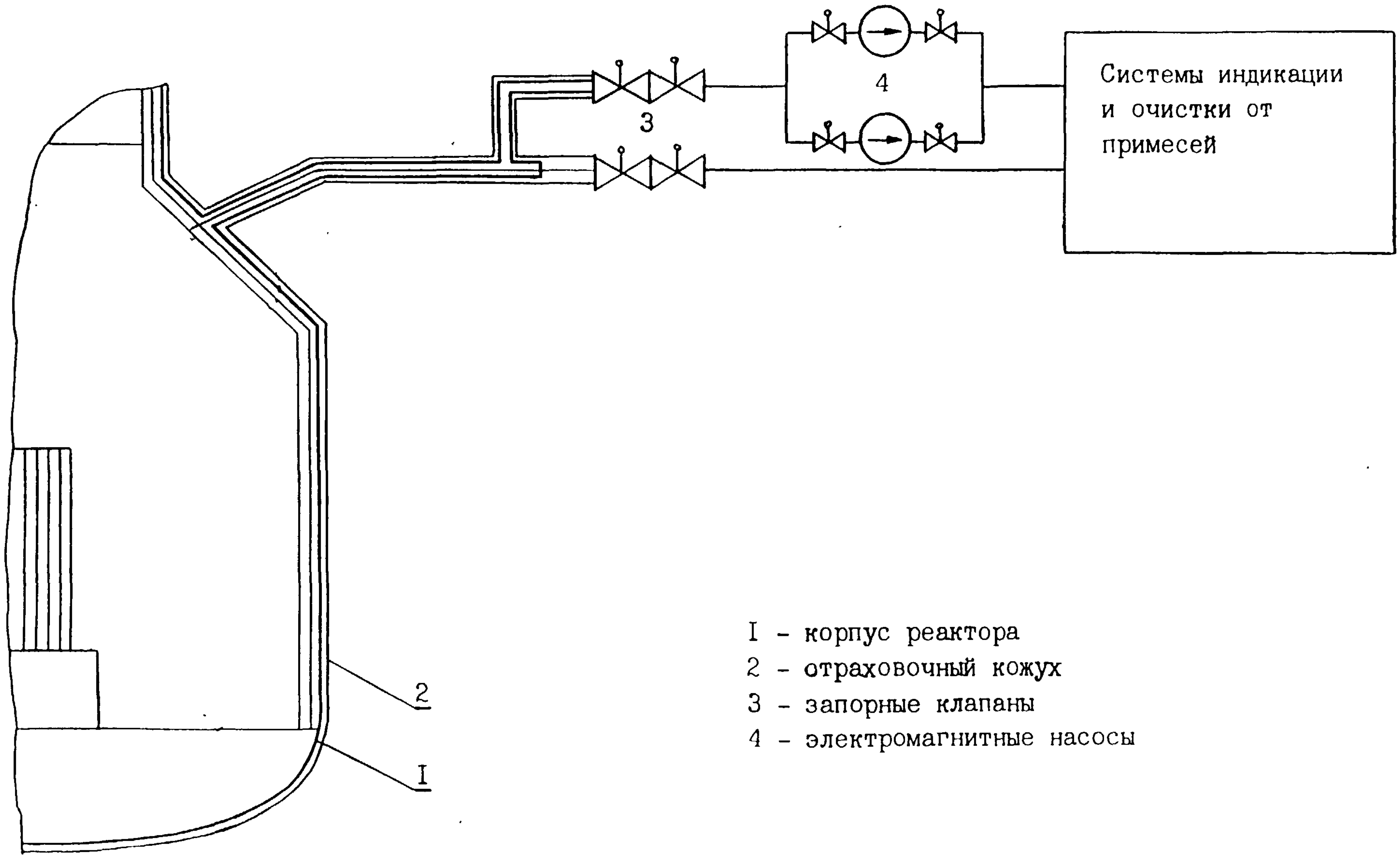
Примером такой ситуации может быть горение натрия, истечение которого сопровождается образованием капель. В частности, исследования показали, что наиболее тяжелым является случай, когда вертикальная струя натрия распадается вследствие удара о потолок помещения или другую горизонтальную преграду, расположенную над местом течи. При этом эффекты горения оказываются значительными.

Подобная ситуация произошла на солнечной станции в Альмерии (Испания) в 1986 году, в которой в качестве теплоносителя использовался натрий [7]. По последующим оценкам всего вытекло 13 - 15 м^3 натрия. Течь продолжалась не менее 15 минут. Это был самый большой (как по масштабам, так и по последствиям) из натриевых пожаров, известных до сих пор. По проведенным впоследствии



Параметр	Натрий	Бензин
Скорость горения	35 кг/м² ч	120 кг/м² ч
Теплота горения (теплотворная способность)	10 000 кДж/кг	44 000 кДж/кг
Скрытая теплота испарения	3 800 кДж/кг	350 кДж/кг
Пламя	Пламя отсутствует	Высота пламени несколько метров
Температура в 80 см от поверхности	Меньше 100 °C	800 °C
Расход воздуха на горение	1 м³/ кг	10 м³/ кг

Рис.1 . Сравнение параметров горения натрия и бензина.



I - корпус реактора
2 - отраховочный кожух
3 - запорные клапаны
4 - электромагнитные насосы

Рис.2 . Схема подсоединения вспомогательных систем 1 контура.

косвенным оценкам температура в зоне горения (распыла) натрия достигала приблизительно 1000 °C. Вне этой зоны температура была значительно ниже.

Подобные ситуации не характерны для инцидентов с течью натрия при нормальных экплуатационных режимах натриевых систем быстрых реакторов. Развитие малой течи натрия в большую, связанное с катастрофическим разрушением системы, происходит достаточно медленно. Малые течи натрия обнаруживаются системами контроля, что позволяет предотвратить их развитие. Крупные течи с разбрызгиванием возможны при ремонтных работах, в основном как следствие грубых ошибок персонала. По-видимому, их нужно анализировать отдельно от рассматриваемых в анализе безопасности проектных и запроектных аварий, т.к. вероятности их реализации при ремонтах и при нормальной эксплуатации резко отличаются между собой.

Изложенное выше позволяет сделать следующие выводы:

- натриевый пожар по выделяемым токсическим веществам при аналогичных условиях не превышает опасность пожара других веществ, применяемых на АЭС;
- тепловые эффекты натриевого пожара, как правило, существенно ниже, чем обычного пожара и только при истечении с образованием капель приближается к нему.

3. КОМПОНОВКА СИСТЕМ БЫСТРОГО РЕАКТОРА

Схема теплоотвода быстрого реактора трехконтурная, т.е. между первым и пароводяным контурами располагается промежуточный натриевый контур. Современные проекты быстрых реакторов, как правило, предусматривают так называемую интегральную компоновку 1-го контура. При этом почти весь радиоактивный натрий размещается в баке реактора - сосуде с двойными стенками, пространство между которыми заполнено инертным газом. Это исключает горение натрия.

Все натриевые системы расположены в отдельных боксах, где не допускается постоянное пребывание персонала. Помещения с радиоактивным натрием, исходя из требований радиационной защиты, имеют железобетонные стены толщиной 1,5 - 2 м . В проектах предусматривается защита бетонных строительных конструкций от воздействия натрия и тепловых эффектов путем создания облицовки стен и покрытия их теплоизоляцией. Многочисленными расчетами и экспериментами было показано, что течь и горение натрия не приводят к разрушению строительных конструкций.

В проектах быстрых реакторов предусматривается герметизация помещений с натрием. К ним не примыкают непосредственно помещения с постоянным пребыванием персонала, тем более с защитными, управляющими или обеспечивающими системами безопасности. Вентиляция помещений с натрием осуществляется отдельными вентиляционными системами, которые никак не связаны с вентиляцией помещений с постоянным пребыванием персонала и с помещениями, где размещены системы безопасности.

4. СИСТЕМЫ ПОЖАРОТУШЕНИЯ НАТРИЯ

Для обнаружения возможных течей и горения натрия, а также для последующего его тушения используются достаточно простые и надежные устройства. Эти системы позволяют надежно обнаруживать факт течи и горения натрия. Пожаротушение натрия обеспечивается в первую очередь пассивными средствами, основанными на эффекте самотушения [8]. Используется герметизация помещений, что предотвращает натекание кислорода воздуха извне и обеспечивает снижение концентрации кислорода до огнеопасного значения. Применяются системы сливного пожаротушения, основным элементом которого являются емкости из нержавеющей стали, расположенные под технологическими помещениями, где в принципе возможны крупные выливы теплоносителя. Вылившийся натрий расплавляет легкоплавкую мембрану, закрывающую дренажную линию, и самотеком сливается в емкости, где из-за недостатка кислорода гаснет. В других случаях используется система поддонов самотушения. Наконец, возможно применение ручных (огнетушители) или стационарных систем порошкового тушения. Отечественная и мировая практика показала достаточность этих мер.

5. ВЕРОЯТНОСТЬ КРУПНОЙ ТЕЧИ НАТРИЯ [9]

Ограждения помещений, в которых размещаются натриевые контуры, представляют собой локализующие системы безопасности. Развитие натриевого пожара, приводящее к угрозе для безопасности реактора, может быть связано с их разрушением (отказом). Как уже сказано, они расчитаны на крупные течи натрия и могут быть повреждены только при наличии скрытого отказа (например, крупных трещин). Поэтому для опасного развития натриевого пожара необходимо наличие следующих событий:

- крупный (полный) разрыв натриевого трубопровода;
- отказ систем борьбы с натриевым пожаром;
- разрушение строительных контструкций.

Тем самым мы приходим к сценарию запроектной аварии.

При анализе запроектных аварий рассматриеваемого типа принимается полный разрыв натриевого трубопровода, хотя прочностные характеристики металла и условия его работы таковы, что внутренних причин для подобных разрушений не существует. Разрушение полным сечением может только в результате развития начального малого дефекта и отсутствия контроля за протеканием этого процесса.

Наибольший интерес с точки зрения обеспечения безопасности АЭС вызывает рассмотрение аварии, связанной с течью радиоактивного натрия первого контура. Предполагается, что подобная течь возможна на трубопроводе вспомогательной системы первого контура, не имеющего страховочного кожуха. На рис. 2. показана схема подключения этой системы к баку реактора с интегральной компоновкой первого контура.

Разгерметизация натриевого трубопровода полным сечением как исходное событие запроектной аварии возможна только в случае ряда отказов в системах безопасности. На рис. 3. представлено соответствующее дерево отказов.

С учетом особенностей технологической схемы вспомогательных систем первого контура (рис.2), принятого дерева отказов систем безопасности и их характеристик надежности была рассчитана вероятность исходного события рассматриваемой аварии, которая оказалась равной 10^{-5} 1/реактор-год.

После разрыва трубопровода и излива натрия в бокс на дальнейшее развитие аварии будет влиять только работа пассивных устройств, к которым относятся:

- клапан избыточного давления, отсекающий приточную вентиляцию при повышении давления в боксе. Однако поскольку повышение давления возможно только на начальной стадии процесса, то срабатывание клапана не учитывается (идет в запас расчета).

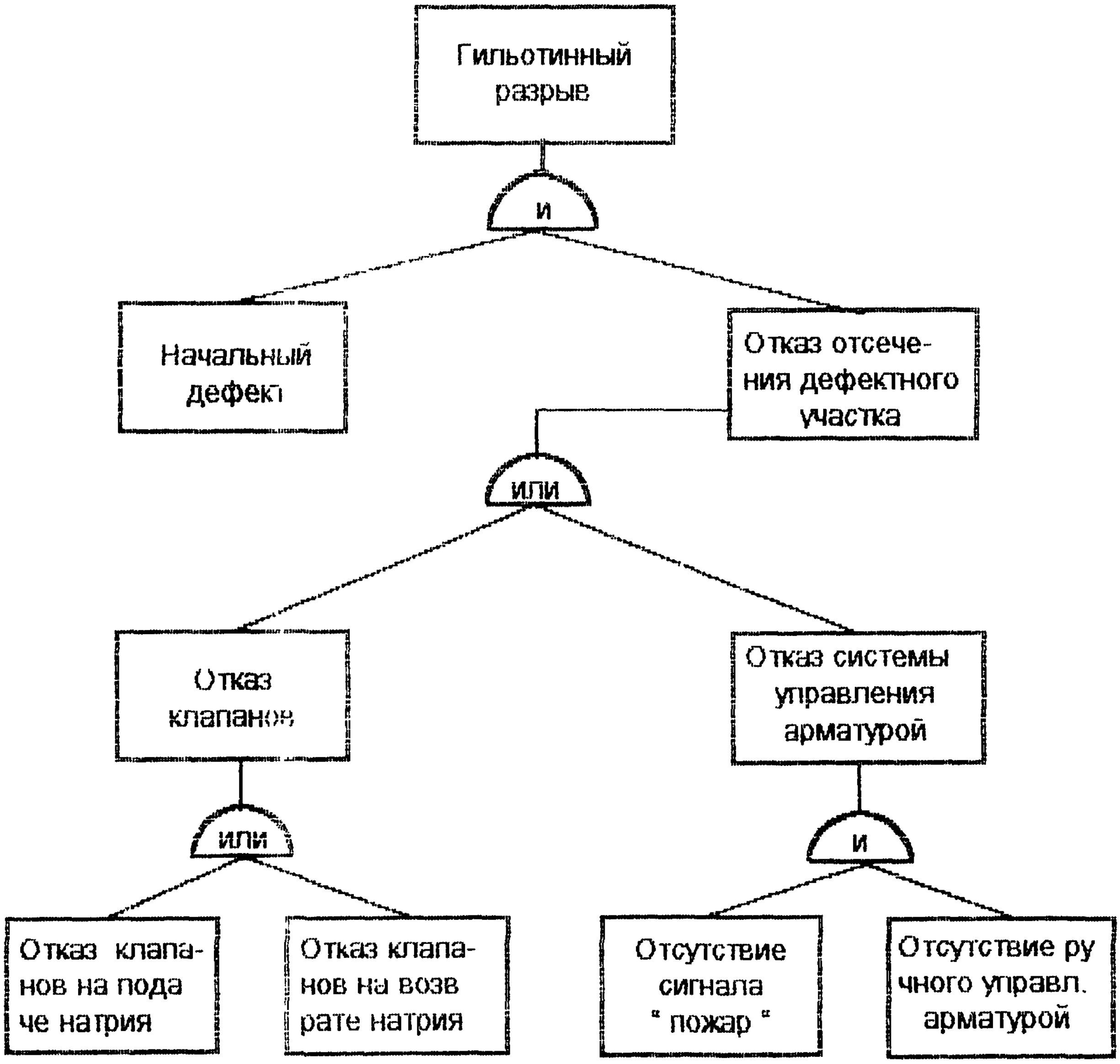


Рис. 3. Дерево отказов, приводящих к гильотинному разрыву натриевого трубопровода вспомогательной системы первого контура

- пассивные средства пожаротушения (поддоны и расширяющиеся порошковые составы) Отказ пассивной системы имеет вероятность 10^{-4}.

Таким образом после наступления исходного события возможны два пути развития аварии со срабатыванием пассивных средств пожаротушения и с их отказом. Для анализа принят второй путь, т.е. вероятность развития аварии по этому пути равна 10^{-9} 1/реактор год.

Для дальнейшего опасного развития событий необходимы очередные отказы систем безопасности Это приводит к дальнейшему снижению его вероятности. Таким образом мы получаем, что вероятность ухудшения ядерной безопасности реактора в результате натриевого пожара лежит в диапазоне остаточного риска.

6 ЗАКЛЮЧЕНИЕ

а) Для реально осуществимых аварийных ситуаций натриевый пожар не вносит кардинального вклада в общую пожарную опасность АЭС.

б) Опасности натриевого пожара для обслуживающего персонала не превышают опасностей от обычного пожара.

в) Натриевый пожар не создает угрозы для ядерной безопасности реактора.

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PANEL 1

Panel 1

IDENTIFICATION OF DEFICIENCIES IN FIRE SAFETY IN NUCLEAR PLANTS

Chairperson: **D.S. Mowrer** (United States of America)

Members: M. Kaercher (France)
M. Kazarians (United States of America)
S. Lee (Canada)
M. Röwekamp (Germany)

The panel was formed to discuss the most effective means of identifying fire safety deficiencies in nuclear power plants. The panel chairman and members included fire safety consultants and nuclear power plant designers. This discussion addressed three aspects of identifying deficiencies:

(1) Appropriate role and reliability of probabilistic safety analysis (PSA)
(2) Most effective means of identifying deficiencies
(3) Evaluation of human error aspects including manual fire fighting response

Fire PSA

The first question addressed to the panel concerned how the human error aspect of manual fire fighting can be factored into a fire PSA. A panel member who is a specialist in fire risk assessment and PSA methodology development answered that the practical details of fighting the fire, including human error issues, typically are not factored effectively into the fire PSA. These analyses typically make conservative assumptions about the success of fighting the fire. Human error concerning operator actions at the control panel typically are factored into the analysis.

The next question was a clarification of the first: specifically, does the fire PSA assign probabilities to the expected performance of the fire brigade? The PSA specialist on the panel answered that the analyst typically will assign a probability for failure of the brigade to get there on time, which involves having the right equipment with them, not tripping on the way to the fire, etc. This probability is largely determined based on the training aspects of fire fighting, including types of drills and available equipment.

The next topic concerned how the PSA analyst should approach plant managers about making decisions given the uncertainties inherent in a fire PSA. The specialist on the panel pointed out two reasons why these analyses are performed: (1) alternative methods of obtaining the same information as that yielded by a fire PSA can be more expensive than the PSA, and (2) the purpose of the fire PSA is to discover hidden or not-so-obvious chains of events which could happen. There was a certain amount of controversy expressed by the attendees regarding the cost-benefit of PSA methodologies, given the fairly wide uncertainty factors of one to two orders of magnitude. The fire PSA entails a very systematic analysis of what could possibly go wrong, and the nature of this analysis yields a reasonable degree of certainty that all possible scenarios and contingencies have been considered. PSA essentially provides a way to articulate or organize knowledge about the fire risk; it does not add to that knowledge base. Any uncertainty arising from the PSA arises from uncertainty in the knowledge base; decisions are always made in the face of some uncertainty and this is no exception. This topic continued with an observation from the floor that plant managers and engineers may be reluctant to implement recommendations, or even to perform a fire PSA, because of (1) the uncertainties inherent in the fire PSA results; (2) the time and expense of performing a PSA; and (3) the nature of some recommendations arising from the PSA which might introduce only a slight improvement to a system, process or area already operating within acceptable parameters.

Expanding on this topic of how to convince plant managers and engineers of the value of the fire PSA and its recommendations, a question was posed concerning the use of the PSA methodology to improve public confidence in the current state of nuclear technology. The concern was that some decision makers might use the PSA results to justify making no improvements, which seems to have been the case with a decision by ValueJet, based on the results of a PSA, to not put fire detection in the cargo bay of the plane which caught fire and crashed into a Florida swamp two years ago. The answer was that the ValueJet management, maintenance or pilot personnel should have had a better knowledge base from which to provide input into the PSA. Convincing the public of nuclear safety is a cultural rather than a technical issue. The concern expressed by some attendees was that even though the existing database is very limited, PSA techniques are still being used—sometimes inappropriately.

Another cultural issue then was introduced concerning the use in the fire PSA of different numbers (probabilities) among the IAEA Member States. The PSA specialist on the panel suggested that plant personnel providing input to the PSA should be knowledgeable enough about the systems and equipment to know if, for example, their pump is the same as one in a U.S. plant; if so, then U.S. numbers could be used. Another participant from the floor who has used PSA in making decisions at an electric utility recommended that the best application of PSA is as a supplemental tool used in addition to good engineering judgment. Used in this way, PSA recommendations can assist decision makers in allocating limited budgetary resources, and this participant indicated that PSA is used as one of many decision making tools, rather than as a sole source of information. The ensuing discussion focused on what other tools are used when plant management is deciding on improvements. Engineering judgment, supplemented with numerical support, is probably the most prevalent tool.

Identifying Deficiencies

The next major topic to be introduced concerned the identification of fire safety deficiencies in nuclear power plants. The first question asked was whether the visual and field inspection yields more insight than the systematic analysis. A panel member with design, FHA and fire PSA experience answered that general deficiencies are better identified using a systematic analysis, and problems specific to a particular operation are better identified through visual inspection. Another panel member stated that the main safety work gets done in the walkdown and that a very small amount of added benefit comes from fire PSA. He then amended this with the observation that PSA might be a useful tool for analyzing deficiencies, such as those associated with cables, which would not necessarily be obvious during a walkdown. Another panel member pointed out that in his experience, the difference between actual knowledge gained from being in the plant and "paper" knowledge of the plant gained from design drawings and other documents viewed from outside the plant is probably two orders of magnitude of difference.

Manual Fire Fighting

The topic of manual fire fighting was introduced as a reiteration of the question which began the session (how manual fire fighting is factored into the fire PSA). Guidance was sought concerning the time scale acceptable for off-site fire service to respond to a plant emergency beyond the plant brigade capabilities. In addition, the question was posed: should the number of fire groups called in to assist correspond with the number or size of the fires, or with the number of reactors at a site? The answer given was that the presence of too many fire fighters introduces its own problems, for example, because a limited number is allowed to enter into the containment area or because of a limited amount of fire fighting equipment (not to mention space limitations). Given these limitations, in addition to difficulty finding the fire to fight it and travel time for outside assistance, this participant recommended that power plants rely more on passive barriers and fire detection/suppression systems, rather than excessive manual fire fighting resources, to assure fire safety of the plant in the first 10 or 15 minutes after fire detection. Another panel member expanded on this answer by emphasizing that very good coordination, as well as administrative controls, should strongly regulate access to special areas. The control for nuclear safety should always be in the hands of the plant personnel.

This panel member again reiterated the question on how to consider the time for fire brigade response when performing fire PSA. The orders of magnitude for success currently used are internationally accepted but give very rough values. She then agreed with the importance of passive measures and automatic fire protection systems. Another panel member indicated that the French plants rely mainly on passive fire barriers and very efficient detection systems. If manual fire fighting is anticipated for a particular area, then automatic suppression systems are installed to compensate for any limitations on the success of the brigade. Manual fire fighting is used with the aim, mainly, to limit the costs of fire damage, rather than to assure plant fire safety.

Fire PSA Revisited

The next question from the floor reintroduced the topic of fire PSA, and it concerned comparing PSA results from plant to plant. Specifically, generic data gathered from other plants are used to supplement the lack of plant-specific data for a given fire PSA, and those generic data will be based upon certain practices in the source plants. Should the plant performing the fire PSA comply with those practices in order to ensure a similar frequency of fire? The PSA specialist on the panel answered that in the nuclear industry, within a given country the variation from plant to plant is not that huge, but if a certain room in a plant has a feature significantly different from a comparable room in other plants, then generic data should not be used. Another panel member dissented with this opinion, offering his own opinion that plants and specific rooms are much too different to compare, regardless of whether the point of comparison is ignition frequencies or risk results.

Another question was posed about whether the shortcomings of fire PSA are due more to the deficiency of the analyst or methodology or to the unavailability of appropriate data. In other words, should PSA specialists concentrate on better applications and existing methodologies or on a more extensive collection system and use of operational experience? The answer given was that operational experience should always be collected because it forms the true knowledge base, which is continually expanding. What PSA analysts should focus on is the chain of events, and all possible chains of events, which could lead to a fire in a particular area.

Summary

The salient opinions which emerged during this panel discussion were as follows: (1) Given the inherent uncertainty within fire PSA, this methodology should be used only as one of several decision making tools, including probably the most prevalent, engineering judgment. This judgment should be incorporated into the PSA process, especially when generic data and probabilities drawn from other plants are used to assign probabilities. The focus of the PSA should be on the chains of events which could cause a fire. (2) From a theoretical standpoint, a systematic approach such as fire PSA seems to be very effective for identifying general deficiencies within a plant or deficiencies that would not be visible or obvious during an inspection. However, from a practical standpoint, a walkdown/inspection is the most effective means for identifying deficiencies related to particular processes or areas. (3) Given the possibility of human error, one effective use of manual fire fighting is to limit the damage caused by a fire, rather than to ensure nuclear safety. Any outside assistance must be coordinated and the safety of the plant must remain at all times in the hands of the plant personnel.

OPERATIONAL EXPERIENCE AND DATA

(Session 4)

Chairperson

M. RÖWEKAMP
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Key Issues Papers

XA9847513

OPERATIONAL EXPERIENCE AND DATA

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The operational experience of nuclear power plants (NPPs) worldwide has shown that the occurrence of fires is not very frequent, but not improbable. Furthermore, the analysis of NPP fire incidents gives the impression that plant internal fires have the potential to cause damage of safety related equipment and therefore can significantly affect the nuclear plant safety.

The presently available data sources on actual fire events at NPPs have not been properly maintained and are often out of date. In some IAEA member states reporting of fire events is not carried out systematically and/or needs improvement with regard to criteria, scope and format of data reporting. Further practical limitations of the databases result from the existing data sources (e.g. fire reporting systems) being not sufficiently detailed to allow for suitable extrapolation to plant and/or fire compartment specific conditions considered in plant specific analyses.

Discussions with fire safety experts and, in particular with analysts, have shown that even the highly developed event reporting systems for NPPs are not able to give the information the analyst really needs for his assessment. It is nearly impossible to obtain information on minor fire related events and/or on precursor events such as electrical shorts leading to tripped breakers or to blown fuses, overheating conditions of components, etc. This information is often not recorded in detail and the access to the data from plant operational records is often limited to the plant/utility level.

Further deficiencies can be found with regard to the available data to support the assessment of various physical behaviour and effects phenomena.

At the time being, in many IAEA member states in the frame of safety reviews etc. activities with regard to probabilistic safety analyses (PSA) including fire PSA are carried out. Furthermore, approaches for risk oriented performance based fire safety standards and guidelines are being developed in some of the member states. The data used for such fire risk studies have been identified as one of the relevant elements requiring enhancement. A considerable part of the data needed for fire safety assessments and, in particular for fire PSA, are derived from operational experience.

The currently ongoing IAEA activities in the field of fire related operational experience and data are focusing on preparing a technical document on "Collection of Fire Related Data at NPPs". The information provided in this document on the potential use of fire related operational experience is intended to assist the plant/utility staff responsible for reporting of data on fire related events as well as the analysts engaged in the fire safety assessment of NPPs. Furthermore, it could be useful for regulatory authorities in setting up requirements for reporting fire safety related events. The purpose of the document is giving assistance in making informed decisions concerning the scope and format of data reporting and also in giving some advice on database features and on data analysis/processing.

The problems and deficiencies with regard to the currently available data and their application in the frame of deterministic as well as probabilistic fire hazard analyses are well known to the fire safety experts and analysts. For more effective future activities the community of fire safety experts has to decide, which types of mainly experience based data have to be gained for carrying out analyses on a realistic basis. Furthermore, an approach has to be chosen how to make more data available without mentioning plant internal information which still should be limited to plant/utility internal use. Incentives have to be found for the utilities to collect more detailed information on all types of fire and fire related events in NPPs considering the needs of the analysts.

IAEA-SM-345/39

Invited Paper

СОСТОЯНИЕ ДЕЛ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ НА АЭС РОССИИ – СТАТИСТИЧЕСКИЙ АНАЛИЗ ЭКСПЛУАТАЦИОННОГО ОПЫТА

XA9847514

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Москва, Российская Федерация

Abstract-Аннотация

FIRE SAFETY STATUS IN NUCLEAR POWER PLANTS IN RUSSIA: A STATISTICAL ANALYSIS OF OPERATIONAL EXPERIENCE.

This paper summarizes the current situation as regards the organization and implementation of fire safety at nuclear power plants in Russia. The design status of fire-fighting systems in nuclear power plants, and the principal measures for improving overall fire safety are described. A statistical analysis is given of fires in nuclear power plants in Russia over the last six years, as well as examples of fires in nuclear power plants in the USSR which are of major significance as regards plant safety. The main lessons learned, and the measures implemented to improve fire safety are summarized. Investigation of fires and reporting is discussed. Problems with the carrying out of probabilistic safety assessments involving comprehensive assessment of the contribution of fires to the frequency of serious damage to the reactor core are examined. In addition, the improvement of fire-fighting systems is discussed.

СОСТОЯНИЕ ДЕЛ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ НА АЭС РОССИИ – СТАТИСТИЧЕСКИЙ АНАЛИЗ ЭКСПЛУАТАЦИОННОГО ОПЫТА.

В данном докладе обобщается реальное состояние дел по организации и обеспечению пожарной безопасности (ПБ) на АЭС России. Излагаются проектное состояние систем пожаротушения на АЭС и основные мероприятия, направленные на повышение ПБ в целом. Приводится статистический анализ пожаров на АЭС России за последние 6 лет, приводятся также примеры происшедших пожаров на АЭС СССР как наиболее значимых для безопасности АЭС. Обобщаются основные уроки и меры по совершенствованию ПБ. Уделяется внимание вопросам расследования пожаров и отчетности. Излагаются вопросы проведения вероятностного анализа безопасности как комплексной оценки по вкладу пожаров в частоту тяжелого повреждения активной зоны реактора. Уделяется также внимание вопросам усовершенствования систем пожаротушения.

1.ОБЩАЯ ЧАСТЬ

1.1 Организация работ по обеспечению пожарной безопасности на АЭС.

В соответствии с действующими нормативными документами Минатома России ответственность за противопожарное состояние и своевременное выполнение мероприятий на АЭС возложена на руководителей АЭС. Все производственные здания и сооружения, административные корпуса, помещения складов, территория и пр. закреплены за структурными подразделениями (цехами) АЭС.

В соответствии с распорядительными документами министерства на всех АЭС разработаны планы мероприятий по повышению ПБ на 1995-2000 годы. Составляются также годовые графики выполнения противопожарных мероприятий. Ежегодно областными структурами Государственной противопожарной службы Министерства внутренних дел Росси (ГПС МВД РФ) проводятся инспекции противопожарного состояния АЭС, в результатом чего администрации АЭС выдаются предписывающие документы для устранения выявленных замечаний, на основании чего на АЭС издаются распорядительные документы, в которых назначаются сроки и ответственные за выполнение мероприятий.

Перед каждым планово-предупредительным ремонтом (ППР) Госпожнадзором России (МВД РФ) проводятся комиссионные проверки противопожарного состояния блоков АЭС, результаты которых учитываются ГАН РФ при выдачи разрешения на пуск конкретного блока АЭС после ремонта.

На всех АЭС организованы инспекции ведомственного контроля за обеспечением ПБ, которая проводит обследования состояния противопожарного оборудования. технической документации, осуществляет контроль за выполнением противопожарных мероприятий.

На АЭС организованы также пожарно-технические комиссии (ПТК) из представителей административного пресонала, которые рассматривают документы по реконструкции систем противопожарной защиты и ПБ в целом на АЭС, периодически проводят осмотры зданий и сооружений, анализируют случаи возникновений пожаров.

На АЭС еженедельно проводятся оперативные совещания по вопросам ПБ и по выполнению противопожарных мероприятий. Организованы также добровольные пожарные дружины (до 600 чел. на кождой АЭС).

На энергоблоках АЭС разработаны оперативные планы и карты пожаротушения, в которых расписаны действия оперативного персонала при возникновении пожаров (до прибытия специальных пожарных подразделений).

В соответствии с программами подготовки персонала по ПБ проводятся учебно-тренировочные занятия совместно с пожарными подразделениями.

На АЭС введен также противопожарный режим (порядок оформления огневых работ, курение и пр.).

Результатом работы различных вышеуказанных структур только в 1996 году было выявлено на АЭС России 19300 нарушений требований нормативной документации по ПБ.

В техническом плане состояние ПБ на действующих и остановленых для снятия с эксплуатации АЭС в настоящее время выглядит следующим образом.

На всех блоках АЭС все кабельные помещения, главные трансформаторы и маслосистемы оборудованы автоматическими установками пожаротушения, которые постоянно находятся в готовности к включению в автоматическом режиме. В машинных залах установлены системы пожарного водопровода высокого давления для подачи воды на лафетные стволы для орошения ферм и колонн машзалов, смонтированы также сухие трубы. На всех АЭС имеется система пожарного водопровода, охватывающая все здания и сооружения, с установленными пожарными гидрантами, пожарными кранами, рукавами и стволами. Имеются сифонные колодцы и пирсы для забора воды пожарной автотехникой. Все помещения на АЭС оснащены первичными средствами пожаротушения. В помещениях контроля и управления блоками АЭС, в помещениях с электронной аппаратурой и в ряде других важных для безопасности помещениях АЭС смонтирована автоматическая пожарная сигнализация и установки газового пожаротушения.

В целях улучшения противопожарной защиты с 1989 года по настоящее время выполнены основные мероприятия, направленные на повышение пожарной безопасности, в частности:

- выполнены огнезащитные уплотнения кабельных проходок в кабелных помещениях;

- завершены работы по покрытию кабельных трасс огнезащитным составом;

- разделены кабельные потоки разных систем безопасности;

- выполнен подпор воздуха в помещениях управления блоками АЭС и автоматизированных систем управления;

- проведена реконструкция дверей и перегородок в кабельных помещениях, выполнено разделение протяженных корридоров и кабелных тоннелей на отсеки и пр.

Часть запланированных мероприятий не было выполнено, в основном, из-за отсутствия должного финансирования.

1.2. Статистический анализ пожаров на АЭС России.

В настоящее время в России действует 29 энергоблоков (9 АЭС) с реакторными установками (РУ) различных типов с установленной электрической мощностью 21242 Мвт (12% от установленной мощности всех электростанций

России), 4 блока остановлены для снятия их с эксплуатации (с наличием на площадках отработавшего ядерного топлива). Владельцем ядерных установок является Минатом России, который образовал 2 эксплуатирующие организации: концерн "Росэнергоатом" (8 АЭС, 25 блоков) и "Ленинградская АЭС" (1 АЭС, 4 блока).

Как известно, построенные до 1994 года АЭС в России, проектировались в соответствии с действующими на тот период общепромышленными нормами и правилами, не отвечающих современным требованиям нормативных документов.

Мировая статистика происшедших пожаров на АЭС подтверждает, что они могут возникнуть из-за незначительных отказов электрооборудования или из-за несоблюдений требований противопожарного режима и могут привести к тяжелейшим последствиям, вызвать отказы систем безопасности (СБ), нарушить отвод остаточного тепловыделения РУ и привести к большим человеческим и материальным потерям.

Значимые для безопасности пожары на Белоярской АЭС (1978г.), на Армянской АЭС (1982г.), на Чернобыльской АЭС (1991г.) и на ряде зарубежных АЭС являются подтверждением вышесказанного.

Характерными примерами могут служить случаи происшедших пожаров на АЭС бывшего СССР.

На блоке 2 Белоярской АЭС (АМБ-160) в последний день старого 1978 года в результате разгерметизации маслопровода турбогенератора возник пожар в машинном зале и распространился по кабельным трассам. К моменту прибытия пожарной команды (через 6 минут после сообщения) перекрытие на площадью в 960 м^2 машзала уже обрушилось. Пожаром были повреждены кабели и электрические панели на отметке 12,3 и 16,4, а также панели в помещении управления блоком (БЩУ-2) на отметке 20,0, что привело к потере контроля реакторной установкой. Ликвидация пожара продолжалась 22 часа (при температуре наружного воздуха -47С° . Пзже машзал был восстановлен, были заменены выгоревшие кабели и блок 2 проработал до 1989 года, после чего был остановлен для снятия с эксплуатации.

На блоке 1 Армянской АЭС (ВВЭР-440) загорание произошло в насосной станции подъема воды (расположенной в мащзале) в результате короткого замыкания на клеммах двигателя насоса. Огонь быстро распространился по кабельным трассам машзала до отметки 9,6, в результате чего произошли массовые короткие замыкания в силовых и контрольных кабелях, вызвавшие взрыв с воспламенением трансформатора собственных нужд, пожар на турбогенераторе и маслобаках. В результате пожара выгорели кабели на площади 400м^2, было повреждено оборудование комплексных распредустройств (КРУ-6кВ) в машзале, что привело к потере контроля реакторной установкой в течение 8 часов. Ликвидация пожара длилась в течение 5 часов.

Энергоблок 2 Чернобыльской АЭС (РБМК-1000) находился в режиме подъема мощности (N=425 Мвт) после капитального ремонта, в работе находился один из двух турбогенераторов (ТГ). Произошло короткое замыкание в кабеле управления работой высоковольного воздушного выключателя, в результате чего произошло его 3-х кратное включение с подачей напряжения на второй (отключенный) генератор, что привело к термодинамическим воздействиям на ротор генератора (непроектный режим асинхронного двигателя), появлению вибрации на ТГ, выбросу и воспламенению водорода (водородное охлаждение ротора генератора) и разливу масла из подшипников генератора. Возник пожар в машинном зале. Реактор был аварийно остановлен оператором. Через 5 минут оперативный персонал смены приступил к тушению пожара на ТГ с помощью лафетных стволов, в это же время прибыла пожарная команда. Были применены все имеющиеся средства пожаротушения в машинном зале блока, но пожар был настолько сильным, что через 20 минут после его возникновения обрушилась кровля машинного зала. В результате происшествия вышло из строя оборудование важное для безопасности, в частности, по общей причине, было потеряно управление питательными насосами (расположенных в машзале), что привело к резкому снижению уровней питательной воды в барабанах-сепараторах (БС). Позже была организована подпитка циркуляционного контура с помощью насосов докачки конденсата и насосов гидроуплотнений, уровни в БС были восстановлены. Ликвидация пожара продолжалась в течение 5 часов. В результате нанесенного большого материального ущерба блок 2 в эксплуатацию больше не вводился.

Таким образом возникшее короткое замыкание в контрольном кабеле привело к тяжелейшему пожару и выводу из строя энергоблока мощностью 1000 Мвт.

Эти и другие примеры говорят о многих причинах возникновения значимых для безопасности пожаров. Во-первых - о недостатках проекта в части выполнения главной электрической схемы открытых распределительных устройств (ОРУ). Во-вторых - о применении горючих кабелей, масел, водородного охлаждения генераторов. В-третьих, проектные металлические конструкции в машзалах АЭС первого поколения не были расчитаны на их длительную огнестойкость. В проектной документации эти (и другие) вопросы не были проанализированы должным образом.

По данным Всемирной организации по эксплуатации АЭС, основные эксплуатационные показатели (количество автоматических аварийных остановов РУ, коллективная доза облучения персонала, готовность систем безопасности, количество пожаров и пр.) большинства АЭС России, на которых были внедрены мероприятия по повышению ПБ, находятся на уровне зарубежных АЭС.

Вместе с тем, в ближайшие годы большинство энергоблоков АЭС России выработают свой эксплуатационный ресурс (Таблица I), а их модернизация и

ТАБЛИЦА I. ДЕЙСТВУЮЩИЕ И ОСТАНОВЛЕННЫЕ БЛОКИ АЭС РОССИИ

№ п/п	Название АЭС	Блок	Тип реактора	Мощность (Мвт эл.)	Год ввода в экспл.	Проектн. срок оконч. эксплуат.
1.	Белоярская	1	АМБ	100	1963	1980*
		2	АМБ	160	1967	1989*
		3	БН-600	600	1980	2010
2.	Билибинская	1	ЭГП	12	1974	2004
		2	ЭГП	12	1974	2004
		3	ЭГП	12	1975	2005
		4	ЭГП	12	1976	2006
3.	Балаковская	1	ВВЭР-1000	1000	1985	2015
		2	ВВЭР-1000	1000	1987	2017
		3	ВВЭР-1000	1000	1988	2018
		4	ВВЭР-1000	1000	1993	2023
4.	Калининская	1	ВВЭР-1000	1000	1984	2014
		2	ВВЭР-1000	1000	1986	2016
5.	Кольская	1	ВВЭР-440	440	1973	2003
		2	ВВЭР-440	440	1974	2004
		3	ВВЭР-440	440	1981	2011
		4	ВВЭР-440	440	1984	2014
6.	Курская	1	РБМК-1000	1000	1976	2006
		2	РБМК-1000	1000	1978	2008
		3	РБМК-1000	1000	1983	2013
		4	РБМК-1000	1000	1985	2015
7.	Ленинградская	1	РБМК-1000	1000	1973	2003
		2	РБМК-1000	1000	1975	2005
		3	РБМК-1000	1000	1979	2009
		4	РБМК-1000	1000	1981	2011
8.	Нововоронежская	1	В-1	210	1964	1984*
		2	В-3	365	1969	1990*
		3	ВВЭР-440	440	1971	2001
		4	ВВЭР-440	440	1972	2002
		5	ВВЭР-1000	1000	1980	2010
9.	Смолениская	1	РБМК-1000	1000	1982	2012
		2	РБМК-1000	1000	1985	2015
		3	РБМК-1000	1000	1990	2020

где: * - остановленные блоки для снятия с эксплуатации.

ТАБЛИЦА II. СТАТИСТИКА ПОЖАРОВ НА АЭС РОССИИ ЗА 1991-1996гг.

АЭС/гг.	1991г.	1992г.	1993г.	1994г.	1995г.	1996г.	Всего:
Балаковская	4	5	3	0	0	0	12
Белоярская	0	1	0	1	0	0	2
Билибинская	3	0	0	0	0	0	3
Калининская	0	1	0	0	0	2	3
Кольская	3	0	1	2	0	0	6
Курская	0	2	1	2	4	1	10
Нововоронежская	0	0	0	1	5	0	6
Ленинградская	1	0	0	0	1	0	2
Смоленская	3	1	1	1	2	0	8
Всего:	14	10	6	7	12	3	52

реконструкция в условиях ограниченного финансирования ведется недостаточно активно, в связи с чем на АЭС возрастает вероятность возникновения аварий и пожаров. Ниже (Таблица II) приведена статистика пожаров на АЭС России за последние 6 лет, из которой видна тенденция снижения количества пожаров (1991 год - 12, 1996 год - 3). Приблизительно во столько же раз (в 1991-1996гг.) снижено общее количество нарушений в работе АЭС, а также сниженв их тяжесть (по международной шкале оценки событий INES).

В нормативных документах МВД РФ нет требований к категорированию (к статусу) пожаров. На АЭС России любое возгорание считается и учитывается как пожар, считается, что неопасных пожаров нет. Естественно, что отсутствие категорирования пожаров затрудняет проведение сравнительного статистического анализа происшедших пожаров, например, с западноевропейскими АЭС или АЭС США.

Следует также отметить, что после пожара в машинном зале блока 2 Чернобыльской АЭС до настоящего времени не было серьезных пожаров, следовательно не проводились и их серьезные анализы.

Тем не менее, статистика зарегистрированных на АЭС России нарушений требований ПБ за последние годы вызывает озабоченность государственных регулирующих органов, так как любое выявленное нарушение противопожарных требований может привести к реальному пожару, в том числе к значимому для безопасности АЭС. В основном, приведенная статистика выявленных нарушений не относится к системам важным для безопасности. Например, в 1995 году из ряда выявленных нарушений только 2,4% нарушений относятся к помещениям, где расположены системы, важные для безопасности.

Наибольшее количество пожаров за последние 6 лет произошло на Балаковской АЭС (4х1000 Мвт, ВВЭР-1000), Курской АЭС (4х1000 Мвт, РБМК-1000) и Смоленской АЭС (3х1000 Мвт, РБМК-1000) в количествах 12, 10 и 8 случаев соответственно. Из 12 пожаров, происшедших на АЭС России в 1995 году, 2 пожара возникли в помещениях машинных залов, где в значительном количестве находятся системы, важные для безопасности АЭС.

Подавляющее большинство пожаров (в процентном выражении) происходит по причинам нарушения персоналом требований пожарной безопасности (в основном, несоблюдение правил при производстве огневых и пожароопасных работ), из-за неисправности технологического оборудования и нарушения технологического процесса.

1.3. Основные уроки, вытекающие из опыта эксплуатации АЭС и принимаемые меры.

Серьезные пожары на АЭС бывшего СССР, часть примеров которых была приведена выше, послужили основой для переосмысления концептуального подхода к обеспечению пожаробезопасности АЭС. Упор был сделан, в первую очередь, на совершенствование нормативной базы, в результате чего вскоре был разработан новый регулирующий документ “Общие положения обеспечения безопасности атомных станций” (ОПБ-88), в основу которого была положена первая концепция обеспечения безопасности АЭС. Были также введены новые противопожарные нормы проектирования АЭС. На уровне Совета Министров

СССР были разработаны первоочередные мероприятия по пожарной безопасности (СМПБ-88), направленные на замену горючих материалов, покрытие кабельных трасс и несущих металлических колонн в машзале огнезащитным составом, совершенствование самих систем обнаружения пожаров и пожаротушения. К выполнению данных мероприятий были задействованы ряд министерств и ведомств, научно-исследовательские институты, проектно-конструкторские организации и непосредственно АЭС. Часть мероприятий еще находится в стадии выполнения. Была введена новая система подготовки персонала АЭС по пожарной безопасности, были также пересмотрены вопросы организации и профилактики противопожарной защиты. Усилился контроль за обеспечением ПБ со стороны ГАН РФ и ГУПО МВД СССР.

В начале 90-х годов началось сотрудничество по вопросам повышения ПБ на АЭС с зарубежными странами, обладающими развитой ядерной энергетикой.

В настоящее время, основываясь на документы МАГАТЭ и Министерства энергетики США, заканчивается разработка новой концепции противопожарной защиты АЭС и методик выполнения анализа влияния пожаров и их последствий на безопасный останов реакторной установки. При финансовой поддержке зарубежных стран на АЭС России выполняется вероятностный анализ безопасного останова реактора при пожаре. Будут также разработаны вероятностные модели, описывающие поведение энергоблока вследствие пожара, а также возможные пути его приведения в безопасное состояние. Модели будут представлять собой деревья отказов систем АЭС и деревья событий в виде логических диаграмм развития аварий.

1.4 Расследование пожаров и отчетность.

Каждый происшедший на территории АЭС пожар расследуется и учитывается в соответствии с нормативными документами как в системе ГПС МВД РФ так и в системе Минатома России.

Цель служебного расследования состоит в выяснении обстоятельств, непосредственных и коренных причин, условий, способствующих возникновению и развитию пожара, а также последствий и величины нанесенного ущерба.

Сообщение о пожаре, явившимся причиной или следствием нарушения в работе АЭС, передается в Минатом России, ГАН РФ, ГПС МВД Рф и в другие организации в соответствии с нормативным документам “Положение о порядке расследования и учета нарушений в работе Атомных станций” (ПН ПЭ Г-005-12-91) а в случае угрозы безопасности АЭС передача информации министерствам и ведомствам производится в соответствии с “Положением о порядке объявления аварийной обстановки, передачи оперативной информации и организации экстренной помощи атомным станциям в случае радиационно-опасных ситуаций или аварий”.

После ликвидации пожара руководство АЭС оценивает характер пожара и его последствия для уточнения состава комиссии по расследованию инцидента. Во всех случаях расследования пожара назначается комиссия. Пожар, который не был связан с нарушением в работе АЭС расследуется комиссией, назначенной приказом директора АЭС. В состав комиссии включаются инспектор, отвечающий за состояние пожарной безопасности на АЭС, представители от ГПС МВД РФ, ГАН РФ (по их требованию), руководители подразделений АЭС. Председателем комиссии назначается главный инженер АЭС.

Результатом работы комиссии является акт расследования пожара (в некоторых случаях комиссия направляет материалы в прокуратуру для возбуждения уголовного дела), в котором излагаются основные и коренные причины возникновения пожара и предложены мероприятия по улучшению состояния ПБ. В соответствии с актом расследования администрацией АЭС издается распорядительный документ, который устанавливает сроки и ответственность за выполнение противопожарных мероприятий.

Ежегодно администрацией АЭС, по согласованию с местными органами ГПС МВД РФ, редставляются в эксплуатирующие организации отчеты о происшедших за прошедший год пожарах.

Ежеквартально, в соответствии с руководящими документами, территориальные военизированные подразделения МВД РФ, непосредственно ликвидирующие пожары, отчитывается перед областными структурами ГПС МВД РФ, которые в свою очередь дважды в году отчитываются перед своим центральным аппаратом.

В соответствии с “Положением о лицензировании деятельности в области использования атомнаой энергии” для получения лицензии на вид деятельности эксплуатирующая организация, при направлении в ГАН РФ (в составе материалов обосновывающих безопасность АЭС) представляет отчет по противопожарной защите АЭС, который кроме общих положений содержит такие разделы, как:

- организация противопожарной защиты при эксплуатации АЭС (нормативно-техническая база, организационно-профилактическая работа, государственный противопожарный надзор, ведомственный контроль, технические средства пожаротушения, подготовка персонала, пожарная команда и др.);

- состояние работ по анализу влияния пожаров на безопасность АЭС;

- основные мероприятия по повышению ПБ.

Банки данных по пожарам на АЭС ведутся как на районном, так и на областном уровне в системе ГПС МВД РФ. В настоящее время в в центральном аппарате ГПС МВД РФ и в эксплуатирующих организациях создаются централизованные банки данных.

В случаях значимых для безопасности АЭС пожаров, информация о причинах и результатах расследований распространяется эксплуатирующими организациями среди других АЭС в виде информационных писем с соответствующими рекомендациями по недопущению аналогичных инцидентов. Аналогичная информация распространяется областной структурой ГПС МВД РФ среди подразделений системы.

1.5. Усовершенствование систем пожаротушения.

Опыт эксплуатации систем пожаротушения на АЭС показал их как положительную сторону, так и их несовершенство. Последнее относится к так называемым блокам АЭС первого поколения, проекты систем пожаротушения которых разрабатывались в 60-е годы в соответствии с общепромышленными нормами и правилами. На этих блоках в настоящее время ведется так называемый "особый режим" эксплуатации.

Радикальное совершенствование систем пожаротушения проводилось совместно с мероприятиями по повышению пожарной безопасности и проводится в настоящее время в рамках графиков по техперевооружению и модернизации блоков АЭС. В частности, на АЭС:

- проведена реконструкция пожарных водопроводов высокого давления с заменой запорной чугунной арматуры на стальную;

- установлены лафетные стволы в машзалах для охлаждения колонн и ферм при пожаре;

- введены в работу установки водяного пожаротушения в помещениях главных циркнасосов, подпиточных насосов и дизель-генераторных станций;

- реконструированы системы автоматической сигнализации с ориентацией на адресную систему пожарной сигнализации;

- сооружены дополнительные пожарные станции и реконструированы пожарные водоводы;

- сооружены помещения контроля ПБ оаздельно для блоков первого поколения и последующих блоков и ряд других мероприятий, плановое внедрение которых задерживается из-за отсутствия должного финансирования.

ТАБЛИЦА III. КОЛИЧЕСТВО УСТАНОВОК ПОЖАРОТУШЕНИЯ И УСТРОЙСТВ СИГНАЛИЗАЦИИ ПО КАЖДОЙ АЭС

Системы / АЭС	*БАЛ*	*БЕЛ*	*БИЛ*	*КЛН*	*КОЛ*	*КУР*	*НВО*	*СМО*	*ЛЕН*	*Всего:*
Количество установок АТП, из них:	429	5	25	124	230	281	155	282	414	1945
- водяного пожаротушения;	387	3	1	119	135	281	54	282	253	1515
- газового пожаротушения;	42	2	1	5	95	0	0	0	34	178
- пенного пжаротушения;	0	0	23	0	0	0	101	0	61	205
- в том числе смонировано в 1996г.	2	0	0	6	5	0	0	0	24	37
Количеств установок АПС,	96	6	11	27	24	28	54	132	98	476
- в т.ч. смонтированных в 1996г.	1	0	1	2	0	0	0	0	1	5

В таблице III приведено количество установок пожаротушения и устройств сигнализации по каждой АЭС (по состоянию на конец 1996 года).

В 1996 году по сравнению с 1995 годом значительно уменьшилось количество ложных срабатываний установок автоматической пожарной сигнализации (1995г. - 407 случаев, в 1996 году - 203 случая) и автоматических установок пожаротушения с пуском огнегасящего состава (1995г. - 29 случаев, 1996г. - 8 случаев).

Ведется соответствующая работа по совершенствованию систем пожаротушения в рамках Лиссабонской инициативы и по линии Европейского банка развития и реконструкции (ЕБРР). С ЕБРР ГАН РФ ведет работы по организации экспертиз проектных решений, на основании которых выдаются поэтапные разрешения на внедрение оборудования на АЭС, в том числе и систем пожаротушения. В частности, при финансовой поддержке ЕБРР на Ленинградской АЭС (РБМК-1000, 4Х1000МВт) внедряется система водяного автоматического пожаротушения с применением центробежных насосов с дизельными приводами.

Запуск каждого дизельного привода будет осуществляться от индивидуальной аккумуляторной батареи.

Трудно решаются проблемные вопросы с обеспечением эффективными системами пожаротушения в помещениях АСУ ТП и электронных спецсистем. Разработан ряд проектных альтернативных решений, одно из которых состоит в организации локальных участков пожаротушения (на уровне отдельных шкафов, панелей).

Естественно, что эффективная работа систем пожаротушения ожидается в том случае, когда будет завершена реконструкция и модернизация всего пожароопасного оборудования, кабельного хозяйства, замены горючих масел, заменено покрытие кровель, пластиката и пр.

1.6. Применение вероятностных методов для анализа пожарной опасности и оценки последствий пожаров.

В настоящее время существует два основных подхода к оценке влияния пожаров на безопасность АЭС. Один из них предусматривает выполнение вероятностного анализа безопасности (ВАБ). Указанный анализ представляет собой комплексную оценку вклада пожаров на АЭС в частоту тяжелого повреждения активной зоны реактора (ПАЗ) на основе получения оценок частот возникновения пожаров в помещениях основных зданий АЭС и определения их последствий в виде отказов оборудования, важного для безопасности, приводящих к возникновению исходных событий аварий и нарушению функций, необходимых для приведения блока в безопасное состояние при этих событиях.

Другой подход к решению указанной проблемы основан на применении детерминистического метода анализа безопасного останова.

Этот метод характерен системным и единообразным подходом к решению вопросов пожарной безопасности АЭС и позволяет находить простой и эффективный, с точки зрения затрат, способ выбора мер, обеспечивающих наибольшее снижение риска. При этом, несмотря на значительный объем проводимого анализа, планирование последовательности операций всего комплекса исследований дает возможность выявить сложную систему взаимодействий и системные логические связи, такие как расположение электрических кабелей и компонентов схемы станции и одновременно выбрать максимально приемлемый путь устранения уязвимых мест, применяя альтернативные стратегии.

Вместе с тем, этому подходу присущ ряд ограничений. Основными из них являются следующие.

ВАБ в силу комплексности подхода является достаточно глубоким и одновременно системным инструментом анализа пожарной опасности и оценки последствий пожаров. Он, в принципе, позволяет определить полный перечень факторов (отражающих специфические свойства проекта АЭС), в наибольшей степени влияющих на величину частоты тяжелого повреждения активной зоны, а также произвести их ранжировку по данному критерию. Преимуществом ВАБа является также независимость результатов от какой-либо субъективно принятой шкалы оценок последствий, что, напротив, характерно для детерминистического анализа.

Вместе с тем, выполнение ВАБ для пожаров представляет собой достаточно объемную и большую по затратам и времени задачу, включающую, в частности, разработку вероятностной модели, описывающей поведение энергоблока при возникновении исходных событий аварий, являющихся следствием пожаров, а также возможных путей его приведения в безопасное состояние. Одной из задач пожарного ВАБ является разработка баз данных по частотам пожаров в помещениях, а также по вероятностным характеристикам надежности элементов АЭС при пожарах, которые зависят от показателей огнестойкости оборудования и тепловых нагрузок, возникающих в процессе пожара. Использование при проведении ВАБ не отвечающих объекту анализа исходных данных может существенно исказить как абсолютные так и относительные количественные его результаты.

Наиболее адекватная и эффективная оценка влияния пожаров на безопасность АЭС может быть получена путем разумного сочетания обоих упомянутых подходов.

Такой подход наиболее целесообразен, когда уже имеются результаты выполненного ранее ВАБ первого уровня для внутренних исходных событий аварий. В данном случае принимается, что задачи разработки вероятностных моделей и создания базы данных по показателям надежности компонентов АЭС, не относящиеся к пожарам решены при проведении указанного ВАБ. Поэтому основной объем работ в рамках пожарного анализа распределяется следующим образом.

В его детерминистической части решаются задачи определения наиболее важных свойств проекта, влияющих на его пожарную безопасность, выполняется анализ помещений, производится отбор пожарных зон. Дополнительно в несколько расширенном по сравнению этим подходом объеме производится определение зависимых от пожара отказов компонентов систем, которые могут быть использованы для приведения блока в безопасное состояние, а также зависимых от пожара исходных событий аварий.

В вероятностной части анализа значительный объем работ приходится на оценку частот пожаров в различных помещениях АЭС, а также разработку вероятностных моделей для отражения реальных сценариев протекания пожаров на блоке, т.е. с учетом оценки вероятностей различных возможных последствий. Отдельными важными задачами также являются создание интерфейсных моделей, необходимых для приведение в соответствие деревьев событий и отказов из ВАБ-1 к спектру последствий, определенных в результате детерминистического анализа и собственно расчет частоты ПАЗ с анализом значимости основных вкладчиков в риск.

Подобный подход применен в настоящее время при анализе безопасного останова в случае пожара энергоблока № 4 Балаковской АЭС с реактором ВВЭР-1000 (проект В-320). Эта работа выполняется специалистами ВНИИАЭС, института Атомэнергопроект и Балаковской АЭС. Ее результаты излагаются в отдельном докладе данного симпозиума.

2. ЗАКЛЮЧЕНИЕ

В течение последних 6-8 лет на АЭС России, в результате полученных уроков после происшедших серьезных пожаров, усилиями структур Минатома РФ, ГПС МВД РФ, ГАН РФ и других организаций проведена существенная работа как по повышению пожаробезопасности, так и по совершенствованию систем пожаротушения действующих и стоящихся АЭС. Особое внимание уделено системе подготовки персонала. За рассматриваемый период времени заметно снижено как общее количество пожаров на АЭС, так и их тяжесть. Исходя из опыта эксплуатации АЭС России можно с достаточной степенью вероятности утверждать, что в настоящее время состояние дел в области обеспечения ПБ приближается к уровню зарубежных АЭС и работа в данном направлении продолжается.

Мероприятия по повышению пожарной безопасности являются основными стратегическими планами АЭС и их выполнение находится под контролем надзорных органов.

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Invited Paper

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DEVELOPMENT OF A FIRE INCIDENT DATABASE FOR THE UNITED STATES NUCLEAR POWER INDUSTRY

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Abstract

The Nuclear Power Industry in the United States has identified a need to develop and maintain a comprehensive fire events database to support anticipated performance-based or risk-based fire protection programs and regulations. These new programs will require accurate information on the frequency, severity and consequences of fire events. Previous attempts to collect fire incident data had been made over the years for other purposes, but it was recognized that the detail and form of the data collected would be insufficient to support the new initiatives. Weaknesses in the earlier efforts included the inability in some cases to obtain fire incidents reports, inconsistent or incomplete information reported, and the inability to easily retrieve, sort, analyze and trend the data. The critical elements identified for the new data collection efforts included a standardized fire incident report form to assure consistent and accurate information, some mechanism to assure that all fire events are reported, and the ability to easily access the data for trending and analysis. In addition, the database would need to be unbiased and viewed as such by outside agencies. A new database is currently being developed that should meet all of these identified need.

BACKGROUND

Several years ago, the nuclear power industry in the United States began considering "performance-based" or "risk-based" approaches to many aspects of nuclear plant operation and equipment maintenance. In addition, the U.S. Nuclear Regulatory Commission (NRC) started reviewing performance-based or risk-informed regulation approaches to replace some of its traditionally prescriptive regulatory methods. Both the utility industry and the NRC identified fire protection programs as a candidate for performance-based consideration.

When the industry began looking at performance based approaches to fire protection issues, it was recognized very early in the process that a comprehensive fire events database would be an essential element needed to support these types of programs. Accurate information on the frequency and severity of fire occurrences would be critical, and the database would need to be viewed as comprehensive and unbiased by regulators and other outside agencies. In addition, the database must be directly available to any utility company or any utility industry organization with a legitimate need to access the data.

There had been several previous efforts to collect utility industry fire incident information; however, each of those was found to have certain drawbacks. The Fire Protection Committee of the Edison Electric Institute (EEI), a U.S. utility industry organization, has been collecting reports on utility fire incidents for many years. These narrative reports pertaining to both nuclear and non-nuclear facilities are voluntarily submitted to promote information sharing, and are discussed by the EEI Fire Protection Committee during its periodic meetings.

However, not all U.S. utilities are members of EEI, and therefore not all fires are reported under this program. In addition, the fire incident information collected by the EEI is not computerized making it difficult to retrieve, sort, and analyze the data.

In the late 1980s, The Electric Power Research Institute (EPRI), another U.S. utility industry organization, was commissioned to develop a computerized database to support industry efforts in performing Individual Plant Examinations of External Events (IPEEE) and plant assessments using the Fire-Induced Vulnerability Evaluation (FIVE) methodology. EPRI developed this database by reviewing fire related Licensee Event Reports that had been submitted to the NRC, reviewing available literature, and by soliciting information from its member utilities. To the extent possible, this effort collected information on fires that occurred in the U.S. nuclear power industry from the mid-1960s through the end of 1992. While the data collected by EPRI was reasonably comprehensive, the analyses done and the products developed by EPRI based on this data were available only to EPRI members who had contributed to funding the project.

In early 1995, representatives of the utility fire protection community approached Nuclear Electric Insurance Limited (NEIL) to determine if NEIL would be interested in participating in the development and ongoing maintenance of a new database for the U.S. nuclear power industry. NEIL is a mutual insurance company that was established by the U.S. nuclear power industry to provide insurance for risks identified in the wake of the 1979 accident at Three Mile Island. A mutual insurance company is simply one that is owned and operated by the companies it insures, its "member companies". NEIL's member companies include every utility company licensed to operate a nuclear power plant in the U.S., and NEIL provides some form of insurance coverage for every operating U.S. nuclear power plant. NEIL's overall loss control program includes periodic in-plant evaluations to assess conditions and to verify compliance with established standards. These evaluations are conducted by NEIL Loss Control Representatives (LCRs) who are fire protection professionals. During each plant evaluation, the LCRs routinely review a plant's fire incident log. Thus, the program provides a built-in mechanism to virtually assure that all future fire incidents will be included in the database. If a fire event occurred that was not reported, it would be identified by the LCR during the normal course of business, and efforts would be initiated to collect the necessary information. With NEIL's involvement at every operating U.S. nuclear plant and its interest in fire protection, it was viewed as a logical participant in the efforts to develop a new database.

FIRE INCIDENT REPORTING FORM AND DATABASE DEVELOPMENT

Some of the initial questions that needed to be addressed included the level of detail needed regarding each fire incident and how the information would be reported and tracked. A task group comprised of utility fire protection professionals was formed to study these issues. One of the first issues that arose was a recognition that a consistent industry definition for a "fire" did not exist. A survey of several nuclear power plants revealed wide variations in how "fire incidents" were defined. The spectrum ranged from one plant that recorded as "fire incidents" all instances where the plant fire brigade was dispatched, regardless of whether or not an actual fire had occurred, to another plant that didn't log events as "fire incidents" unless an actual fire burned for longer than ten minutes. Obviously, in order to obtain consistent and meaningful data, a standard definition was needed. The task group drafted a "fire incident" definition and solicited comments and input from other utility industry fire protection professionals. Ultimately, the following definition of a "fire incident" was adopted:

> *"A reportable fire incident is one which results in the use of manual fire suppression activity; or, results in the activation of an automatic fire suppression system; or, shows visible flame or evidence of prior flaming."*

Reportable fire incidents are not intended to include overheating of equipment, smoked components, steam leaks, false alarms, or unfounded odors. This definition is intentionally broad and is intended to capture all actual fire events, regardless of whether or not they caused damage or were significant from a safety standpoint.

Once the definition of a reportable fire was established, attention then turned to the development of a standardized reporting form. Based on input from the utility industry fire protection community, it was decided that a common reporting form pertaining to both nuclear and non-nuclear electric generating facilities would be useful and less confusing to the industry. It was recognized that information on fire incidents in conventional portions of a plant, such as the turbine building, would be applicable to either type of facility. By collecting information from both nuclear and non-nuclear plants, the utility industry would benefit from having one central location in which to store and from which to query fire incident data.

The primary goal was to make the reporting form comprehensive yet easy to use. Everyone involved agreed that the likelihood of having reports voluntarily submitted would increase significantly if the report form was user-friendly and uncomplicated. After identifying the critical data elements, a reporting form modeled after the example forms contained in National Fire Protection Association (NFPA) Standard 902 "Fire Reporting Field Incident Guide" was developed. The reporting form consists of a single page report intended to collect basic information about the fire, and a multiple page nuclear supplement intended to gather more detailed information about fires occurring in nuclear power plants. To help maintain consistency, a numerical coding legend was established for each option that could be entered into a given field. These numerical codings also facilitated the computerization of the form using commercially available software that will allow reports to be filled out and submitted electronically.

With the critical data elements identified, development began on the database itself. Key criteria included the ability to easily retrieve, sort and trend the data. The ultimate goal was to make the database directly and easily available to any legitimate user while protecting it against inappropriate uses. Once again, commercially available software was used to construct a database that will allow remote users to electronically submit incident reports and also to conduct queries. Several standard queries and reports are being developed; however, a user will eventually have the capability to construct customized queries and obtain customized reports. It is intended that access to the database will be available via a secure web site on the Internet. Users will be issued a password that will allow queries to be made and reports to be obtained based on whatever criteria they desire. However, for data security purposes, it will not be possible for a user to download the entire database.

DATA COLLECTION EFFORTS

Data collection for new fire incidents began in early 1996. As noted above, an earlier database developed by the Electric Power Research Institute included data through the end of 1992, and efforts were also made to collect and reconstruct data for the period 1993 - 1995. NEIL's LCRs began collecting information on past fire incidents during routine plant evaluations. Copies of the new fire incident report form were distributed to nuclear plant fire protection personnel, and presentations explaining the new database and report form were conducted at several meetings of industry fire protection organizations. As anticipated with any new program, submittal of fire incident reports directly from the plants has been somewhat slow. NEIL's LCRs continue to identify recent incidents that had not been reported. However, it is anticipated that report submittals will become more routine as the program becomes better known and the electronic submittal process is available to all plants.

CURRENT STATUS

Information on approximately 200 fire incidents that occurred during the period 1993 through mid-1997 have been collected. The reports are currently being reviewed for consistency and completeness, and efforts are underway to fill gaps in information concerning many of these fires. In addition, an agreement was recently reached with the Electric Power Research Institute (EPRI) whereby EPRI will make available all data from their earlier database so that it can be included in the new database. Thus, the new database will represent a reasonably comprehensive collection of information on fires that have occurred in the U.S. nuclear power industry since the mid-1960s.

Software packages have been provided to approximately ten plants for testing of the electronic incident report form and the electronic submittal procedure. In addition, NEIL is in the process of developing an Internet web site where it is anticipated that access to the database will be provided. Users will be able to open an electronic version of the reporting form, enter the necessary information, and submit the report electronically. Querying of the database will also be possible from the web site. It is anticipated that access to the database via the Internet will be available near the end of the first quarter 1998. While NEIL will act as the custodian of the database, it does not intend to trend, interpret or analyze the data for other than its own interests. Interpretation of the data for industry use will be left to industry users or to other industry organizations such as EPRI.

All efforts to date have been directed toward obtaining fire incident data from U.S. nuclear power plants. The reporting form has also been provided to the fire protection group within the Edison Electric Institute, and it is anticipated that in the future, incident reports pertaining to non-nuclear electric generating facility fires will be submitted for inclusion in the database. However, for the non-nuclear plant incidents, all reporting will be strictly voluntary and there will be no mechanism in place to assure accuracy or completeness of the data.

CONCLUSION

The availability of a comprehensive, unbiased fire incident database will be critical to the successful future implementation of performance-based fire protection programs at U.S. nuclear power facilities. While the database currently under development should provide the industry with an essential tool, success will be dependent upon participation by the entire industry. Failure by some to report fire incidents will render the database less comprehensive, and therefore, less useful or effective.

Fire incident information gathered to date confirms the intuitive notion that fires in nuclear power facilities are relatively rare, and those that occur are usually minor and have little or no safety significance. However, a few recent fires have identified certain plant vulnerabilities that had not been previously identified or considered. The collection and sharing of information on these types of incidents can only serve to benefit the entire industry.

In addition to supporting performance-based programs as discussed above, the availability of a comprehensive fire events database could be useful in other areas. Accurate information on fire frequencies and probabilities could be used to make better informed decisions on resource allocations, capital expenditures, etc.

GERMAN DATA FOR RISK BASED FIRE SAFETY ASSESSMENT

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Abstract

Different types of data are necessary to perform risk based fire safety assessments and, in particular, to quantify the fire event tree considering the plant specific conditions. Data on fire barriers, fire detection and extinguishing, including also data on secondary effects of a fire, have to be used for quantifying the potential hazard and damage states.

The existing German database on fires in nuclear power plants (NPPs) is very small. Therefore, in general generic data, mainly from US databases, are used for risk based safety assessments. Due to several differences in the plant design and conditions generic data can only be used as conservative assumptions. World-wide existing generic data on personnel failures in case of fire fighting have only to be adapted to the plant specific conditions inside the NPP to be investigated.

In contrary, unavailabilities of fire barrier elements may differ strongly depending on different standards, testing requirements, etc. In addition, the operational behaviour of active fire protection equipment may vary depending on type and manufacturer.

The necessity for more detailed and for additional plant specific data was the main reason for generating updated German data on the operational behaviour of active fire protection equipment/features in NPPs to support risk based fire safety analyses being recommended to be carried out as an additional tool to deterministic fire hazard analyses in the frame of safety reviews.

The results of these investigations revealed a broader and more realistic database for technical reliability of active fire protection means, but improvements as well as collection of further data are still necessary.

1 INTRODUCTION

In order to perform risk based fire safety assessments, different types of data are necessary to quantify the fire event tree, which can be developed in a general manner but has to be adapted to the plant specific conditions of the relevant fire compartments and areas in the nuclear installation to be investigated. The following types of data are needed:

- fire occurrence frequencies,
- fire spreading parameters,
- unavailabilities of active and passive fire protection measures,
- failure rates for personnel actions in case of fire extinguishing.

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

In Germany there are only few data existing on fires occurring in nuclear power plants (NPPs). Therefore, in general generic US data and to some extent also German data on fire occurrence frequencies are used for risk based fire safety assessments. These data have to be compared to the plant specific conditions. For specific plant areas the generic data can only be used as conservative assumptions because of the differences to the plant specific location and conditions.

At the time being, the procedure for determining fire spreading parameters is a deterministic one. But probabilistic distributions of these parameters are also needed for carrying out quantitative fire safety assessments and are intended to be evaluated in Germany in the long-term.

Data on failure rates for personnel actions in case of fighting fires can be taken generically from all types of fires occurring in nuclear power stations over the world. They only have to be adapted to the plant specific conditions inside the NPP investigated.

In contrary, the data concerning the unavailabilities of fire barriers differ from country to country due to the different fire resistance rating required by national standards and the respective testing procedures for the barrier elements also differing depending on the standards. Generic data, particularly from the US, therefore cannot be used for probabilistic fire considerations. It is possible to take generic German data for fire barrier elements, also from the non-nuclear German industry, as the standards for fire barrier rating and testing are the same as for German NPPs.

In the past, probabilistic considerations in order to evaluate the current fire protection level of a NPP have been used for case-by-case decisions by the utilities and the supervising authorities. However, in the frame of periodic safety reviews of operating NPPs the recently published regulatory guiding documents [1], [2], [3] include a risk based fire safety assessment, providing models, methods and data, which could be used to perform a PSA (probabilistic safety assessment).

2 PROBLEMS WITH THE DIFFERENT TYPES OF DATA NEEDED FOR FIRE PSA

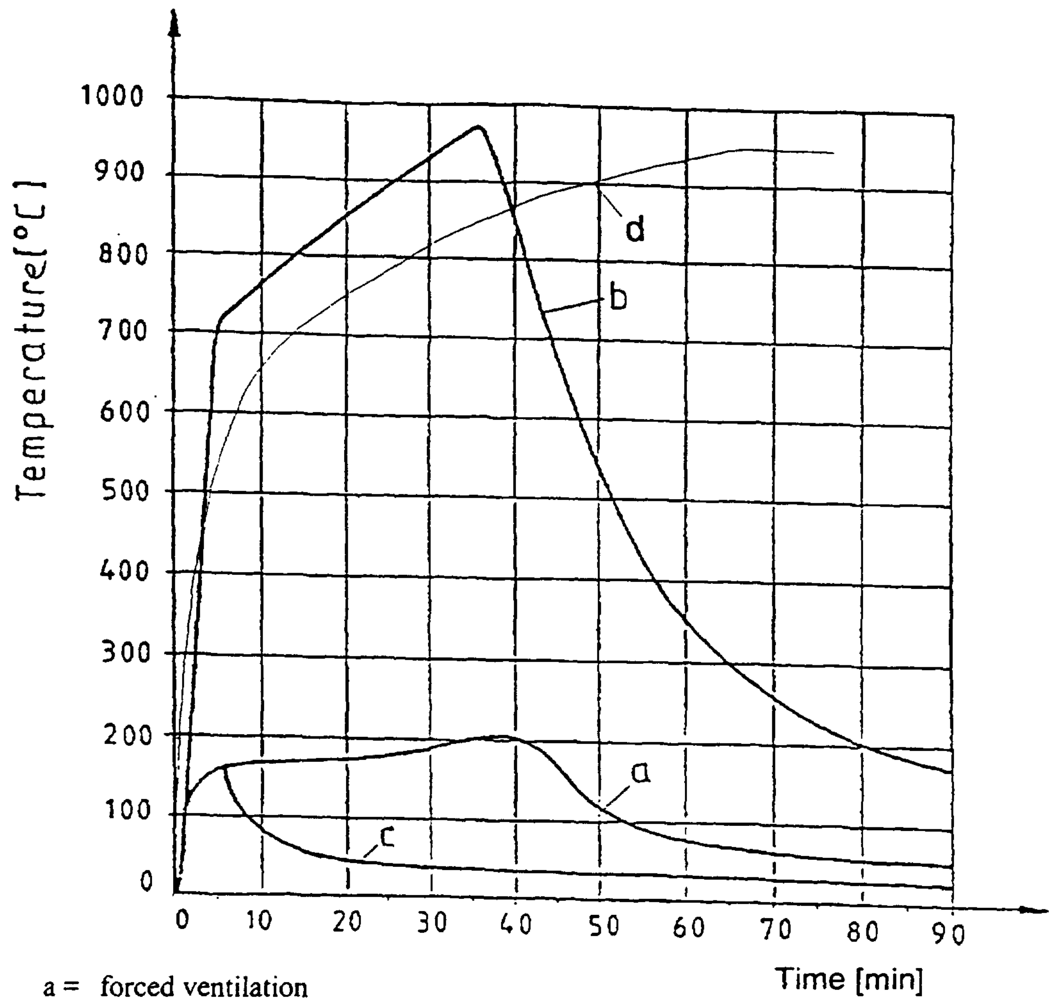
From the operating experience of German nuclear power plants with in total approximately 500 reactor years only 24 fire events were identified by the criteria for obligatory reporting to the licensing and supervisory authorities. Several pilot fires or fires in non-significant plant areas which could give a good indication for the root causes of more significant ones were of course not reported. These 24 fires represent an amount of less than 1 % of all incidents reported. Nevertheless, it is well known that plant internal fires represent significant internal hazards and have to be considered also in PSA studies. The officially available German database of reported fire incidents is too small for use in probabilistic studies. The databases internationally available, such as IRS by OECD and INES by IAEA or the SANDIA fire database, are also very small and to some extent not directly applicable to the specific plant to be analysed. It is therefore necessary to use plant specific information as well as the available generic data and adapt these as far as possible to the plant specific boundary conditions. For this application of generic data engineering judgement and a detailed expert knowledge are indispensable.

After estimation of the fire occurrence frequency for the compartments and areas selected by qualitative and/or quantitative screening the fire spreading in the affected area or compartment and the fire spreading to adjacent compartments or plant areas have to be assessed.

For the analysis of the fire spreading inside the affected fire compartment the fire compartment boundaries, the ventilation conditions and active fire protection measures (fire detection and extinguishing) have to be considered (details are outlined in [2]).

The fire spreading to adjacent compartments can be estimated with different methods with a differing level of conservatism.

The more conservative ones are based on heat transfer calculations considering e.g. a simplified certification procedure for structural fire protection features. By calculating the equivalent fire duration considering the type of combustibles, the fire load density per floor area and the ventilation conditions inside the fire compartment the necessary fire resistance rating of the fire barriers is estimated and compared with the fire rating of the existing barriers.



a = forced ventilation
b = forced ventilation and additional opening
c = forced ventilation switched off after 5 min
d = ISO curve

Figure 1: Time temperature curve variations depending on ventilation (from [15])

The more plant specific and less conservative method is based on an analysis of simulated time-temperature curves of the hot gas layer in the fire compartment in comparison to the standardised time-temperature ISO curve. The heat release rate (energy, integral of the gas temperature T over the fire duration t) of the ISO fire test is compared to that calculated for a real fire under realistic conditions.

Experiments in compartments with realistic NPP specific conditions [4] showed that without any fire extinguishing measures considered the temperature nearly always exceeds the critical temperature of the fire barriers or the safety related equipment inside the compartment for a short time period before decreasing again until the fire is extinguished. Another result of the studies carried out was that for the realistic fire generally this critical temperature T_{crit} is reached at a critical time t_{crit} being not identical with the t_{crit} of the standardised ISO curve (see figure 1). This is considered by the so-called equivalent fire duration ($t_{equivalent}$). Nevertheless, the calculated realistic time-temperature curve varies depending on several parameters for the fire spreading, such as type of the fire load, pyrolis rate/rate of combustion, geometric parameters (e.g. height, volume of the compartment, location of the fire loads etc.), and ventilation parameters. Up to now, the more and more highly sophisticated fire simulation models and codes are only partly available to consider these effects caused by the parameter variations.

Therefore, at the time being, the fire spreading scenarios simulated can only be taken as a more conservative deterministic approach. In the future, all parameters affecting the probability of the critical barrier/equipment failure temperature p (t_{crit}(failure)) have to be implemented in the analyses/models/calculations. It is therefore necessary to estimate probabilistic distributions for all parameters affecting the fire spreading for being able to carry out more realistic fire risk analyses.

Further analyses are necessary to evaluate which of these parameters affecting the fire propagation become relevant at which/what time. It then should be possible to develop more realistic fire scenarios considering most of the parameters affecting the fire propagation as well as secondary fire effects, such as smoke and soot production, overheating of sensitive electric/electronic equipment etc.

3 ACTIVE FIRE PROTECTION MEASURE RELIABILITY DATA FOR FAULT TREE ANALYSES

The operational behaviour of active fire protection measures may vary depending on the type and the manufacturer of the systems and components. Therefore generic data for the technical reliability of such measures can only be considered to some extent. Partly plant specific data are needed. In particular, data from nuclear installations differ from those in non-nuclear ones as gained from the insurance companies because of the more frequent and detailed inspection and maintenance.

This was the main reason to carry out actual investigations for generating updated German plant specific as well as generic data on the operational behaviour of active fire protection features in NPPs in order to support fire PSA activities.

In this context, the operational behaviour of different fire protection measures, such as fire detection systems, stationary fire extinguishing systems, and structural fire protection measures (doors, dampers) in German NPPs was analysed. The analyses were carried out for two German NPPs of different type, a PWR and a BWR, both under operation since the mid eighties for an operating period of about seven years. For those components or systems which belong to the active fire protection means, the reliability data to be estimated were unavailabilities per demand or failure rates λ (t) per hours of plant operation.

For estimating the technical reliability of active fire protection features the plant specific documentation of regular in-service inspections and, in addition, as far as available and assessable, the inspection and maintenance records were analysed. For the analyses several types of documentation containing all technical disturbances as well as deficiencies and failures of systems and components detected during inspections and walk-throughs had to be considered, such as records of periodic tests, regular inspections and maintenance (incl. test and inspection procedures and reports), work and maintenance orders, reports of deviations from normal operation state, and repair reports, if necessary.

For these analyses of technical unavailabilities the human factor was not considered. Results on further investigations on the human influence factor are available from several internationally published reports and have to be taken into account when preparing a scenario specific fault tree diagram for an individual fire risk analysis. Furthermore, it has to be stated that expert knowledge of the plant specific conditions and a well balanced engineering judgement are necessary for the decision to what extent the data can be realistically applied to the plant being analysed.

For the two reference plants analysed in the recent German fire protection features reliability study [6] the plant specific reliability data estimated are given in table 1. A comparison of the generically estimated recent German data with data available from international literature is given in table 2.

The analyses of failures and unavailabilities gave the impression that most of them are single disturbances without relevance for the plant safety. However, some failures with safety significance occurred.

For some of the fire protection features investigated the data base is still too small and has to be expanded.

The comparison of the actual plant specific reliability data for various fire protection features gained from two representative German reference plants with former plant specific data for the same fire protection features and with generic data of the German insurance companies for comparable measures installed in non-nuclear installations showed up deviations.

Table 1: Plant specific technical reliability data estimated for active fire protection features in two German NPPs (from [6])

Active fire protection features	BWR				PWR			
	Inspection period	Scattering factor k	Failure rate $\lambda(t)$ [1/h]	Unavailability per demand	Inspection period	Scattering factor k	Failure rate $\lambda(t)$ [1/h]	Unavailability per demand
Fire alarm panels/boards:								
detection drawers	3m. 1a	3.31	$6.7\cdot10^{-8}$	$1.2\cdot10^{-4}$	3m	7.63	$3.0\cdot10^{-8}$	$7.4\cdot10^{-5}$
detection lines	3m. 1a	3.29	$2.3\cdot10^{-8}$	$4.0\cdot10^{-5}$	3m	7.63	$1.7\cdot10^{-8}$	$4.3\cdot10^{-5}$
Fire detectors:								
automatic	1a	1.25	$1.4\cdot10^{-7}$	$1.3\cdot10^{-3}$	1a	1.75	$4.8\cdot10^{-8}$	$4.2\cdot10^{-4}$
press button	1a	2.55	$1.1\cdot10^{-7}$	$9.4\cdot10^{-4}$	1a	7.63	$3.4\cdot10^{-8}$	$3.3\cdot10^{-4}$
Fire dampers	3m. 1a. 3a	1.13	$2.3\cdot10^{-6}$	$7.1\cdot10^{-3}$	6m	1.93	$6.6\cdot10^{-7}$	$2.9\cdot10^{-3}$
Fire doors	1a	1.36	$4.9\cdot10^{-7}$	$4.3\cdot10^{-3}$	1a	1.44	$7.1\cdot10^{-7}$	$6.3\cdot10^{-3}$
Spraywater deluge systems (dry pipes)								
– total failure of the system	6 m. 1a. 5a	3.30	$.3\cdot10^{-7}$	$9.6\cdot10^{-3}$	3 m. 1a	7.63	$1.3\cdot10^{-7}$	$5.6\cdot10^{-3}$
– failure of automatic actuation	6m. 1a. 5a	1.35	$4.1\cdot10^{-6}$	$1.8\cdot10^{-2}$	3m. 1a	1.26	$1.3\cdot10^{-5}$	$2.9\cdot10^{-2}$
Spraywater deluge systems (wet pipes)	6m. 1a. 5a	2.98	$6.4\cdot10^{-8}$	$3.0\cdot10^{-4}$	-	-	-	-
CO_2 extinguishing systems	6m	7.63	$3.8\cdot10^{-6}$	$1.8\cdot10^{-2}$	-	-	-	-
Halon systems	6m	7.63	$7.6\cdot10^{-6}$	$3.5\cdot10^{-2}$	-	-	-	-
Stationary fire pumps	1m. 1a	7.64	$1.9\cdot10^{-6}$	$1.4\cdot10^{-3}$	1m. 1a	7.64	$1.3\cdot10^{-6}$	$1.2\cdot10^{-3}$
Wall hydrants	6m. 1a	7.64	$3.9\cdot10^{-8}$	$1.9\cdot10^{-4}$	1a	1.75	$8.5\cdot10^{-7}$	$7.4\cdot10^{-3}$

Table 2: **Comparison of the generic reliability data recently estimated for active fire protection features in German nuclear power plants with internationally available data**

Active Fire Protection System	Unavailability per demand estimated for German reference plants		International Nuclear Power Plants								Non-nuclear German Industry		
	BWR	PWR	[7]	[8]	[9]	[10]	[11]	[12]	[13]	[14]	[4]	[16]	[17]
Fire alarm panels/boards:													
– **detection drawers**	1.17E-04[1]	7.36E-05[1]											
– **detection lines**	3.98E-05[1]	4.29E-05[1]											
Automatic fire detectors	1.27E-03[1]	4.22E-04[1]	9.00E-02[1]	-[2]				-[2]			4.00E-03[1]		7.90E-02[1]
Press button detectors	9.43E-04[1]	3.31E-04[1]											
Fire dampers	6.65E-03[1]	2.90E-03[1]											
Smoke extraction dampers	1.55E-02[1]	3.26E-03[1]											
Fire doors	4.25E-03[1]	6.25E-03[1]										≤3.00E-01	
CO_2-Systems	1.81E-02[1]		2.00E-03[1]	1.42E-01[1]	1.16E-01	4.00E-02		4.00E-02	7.00E-02				
Halon systems	3.50E-02[1]			5.36E-02[1]	2.00E-01		5.90E-02	6.00E-02	6.00E-02				
Spraywater deluge systems (dry pipes):			6.30E-03[1]	3.80E-02[1]	4.90E-02				4.00E-02	7.00E-02	6.50E-02		8.00E-02[1]
– ***total failure of the system***	9.90E-04[1]	3.23E-04[1]											
– **failure of automatic actuation**	1.78E-02[1]	2.93E-02[1]		2.64E-02[1]				6.00E-03[1]					
Spraywater deluge systems (wet pipes)	3.15E-04[1]												

[1]= Technical failures per demand

[2]= 8.60E-02 - 2.40E-01 per demand depending on detector type

As a first result the unavailabilities of most of the fire protection measures analysed actually were lower than those gained from former analyses. For some specific measures the unavailabilities were higher. The reason for the deviations should result from more realistic assumptions for the actual investigations on the one hand and from backfitting actions taken as a result of the former analyses in the reference plants.

As another result of the overall analysis differences were found concerning the reliability of fire protection measures between the nuclear and the non-nuclear field. In general, the technical reliability was higher for those measures installed in nuclear power plants. These deviations may result from more frequent and detailed inservice inspections for the equipment in nuclear installations with a highly sophisticated quality assurance and maintenance program as well as from the strongly regulated qualification programmes for the NPP personnel.

4 CONCLUSIONS

The potential of internal fires to simultaneously damage systems and/or components of redundant safety relevant equipment requires that the current fire protection level of operating NPPs is analysed. In this process, especially those NPPs have to be assessed carefully which have been designed and built to earlier standards taking into account the already implemented fire protection improvements.

The review of the current state of fire protection measures and their reliability is also part of the periodic safety reviews which have already been performed for more than half of the operating NPPs in Germany. In the past, these reviews were mainly focusing on deterministic investigations.

However, the recently published regulatory guidance for performing periodic safety reviews, developed by the utilities on a voluntary basis and reviewed by the supervising authorities and their experts also recommend a fire PSA as an additional tool to the deterministic analyses. These procedures require appropriate methods, models and data.

In particular, preparation of the necessary data is a major prerequisite of a risk based fire safety assessment. Therefore, a separate chapter in the PSA document on data presently available in Germany [3] provides material based on national as well as international, mainly US, data. In this document, the recently determined data on the technical reliability of active fire protection measures have been included to provide a broader data base although the use of plant specific data is always preferable.

However, it should be underlined that the necessary data base still has to be improved and to be expanded for several fire protection features. Moreover, the influence of human actions is to be taken carefully in consideration. In particular for German NPPs, the use of US data for the fire occurrence frequencies is insufficient due to the consequential conservative assumptions just at the starting point of the quantitative calculations; on the other hand, the presently available German data do not allow to provide verified data.

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ESTIMATION OF FIRE FREQUENCY FROM PWR OPERATING EXPERIENCE

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Abstract

In the framework of a fire probabilistic safety assessment (Fire PSA), the French Institute for Nuclear Safety and Protection (IPSN) has developed a method for estimating the frequency of fire in a nuclear power plant room. This method is based on the analysis of French Pressurized Water Reactors operating experience. An interesting characteristic of this experience is that the reactors population is homogeneous in terms of design, maintenance and operating procedures.

The method adopted consists in carrying out an in-depth analysis of fire-related incidents. A database has been created including 202 fire events reported in 900 MWe and 1300 MWe reactors from the start of their commercial operation up to the first of March 1994, which represents a cumulated service life of 508 reactor-years. For each reported fire, several data were recorded among which :

- the operating state of the reactor in the stage preceding the fire,
- the building in which the fire broke out,
- the piece of equipment or the human intervention which caused the fire.

Operating experience shows that most fires are initiated by electrical problems (short-circuits, arcing, faulty contacts, etc...) and that human intervention also plays an important role (grinding, cutting, welding, cleaning, etc...)

A list of equipment and of human interventions which proved to be possible fire sources was therefore drawn up. The items of this list were distributed in 19 reference groups defined by taking into account the nature of the potential ignition source (transformers, electrical cabinets, pumps, fans, etc...). The fire frequency assigned to each reference group was figured out using the operating experience informations of the database.

The fire frequency in a room is considered to be made out of two contributions : one due to equipment which is proportional to the number of pieces of equipment from each reference group contained in the room, and a second one which is due to human interventions and assumed to be uniform throughout the reactor. Formulas to assess the fire frequencies in a room, the reactor being in a shutdown state or at power, are then proposed.

To conclude, in the light of initial applications, it appears that the method developed by IPSN is easy to use. Based mainly on the use of French operating experience, which represents a large number of reactor years, it guarantees with an adequate level of confidence that the data obtained will be representative. One original feature of this method is that it takes into account fires breaking out because of human intervention and that it can be applied to power reactors in operation or in shutdown conditions.

1. INTRODUCTION

The French Institute for Nuclear Safety and Protection is performing a fire-probabilistic safety assessment (Fire PSA). In the framework of this study, a method for estimating the frequency of a fire in a given reactor room has been developed. This method is based on the fire related experience gained with the 508,2 reactor-years of operation in the French PWRs.

2. FIRE-RELATED OPERATING EXPERIENCE

2.1 Fire events database

The feedback of French operating experience is based on data available to IPSN. From their first commercial operation up to the first of March 1994, the French 900 MWe and 1300 MWe PWRs accumulated an operating time of 508,2 reactor-years and a total of 279 fires were recorded. These data concern all the fires that occurred in the PWRs which were reported by Electricité de France to the French Safety Authority.

No screening was done on the fire experience feedback as regards the importance of the damages to the equipment. Notably, events limited to smoke production or small fires which went out without intervention are taken into account.

On the other hand, some fire events were excluded based on the following :

- fire events not considered in EdF feedback of operating experience (fires occured before the first criticality or were judged as "not significant", ie outside the nuclear island),
- fire events which occured before the start of commercial operation,
- fire events which occured in buildings outside the scope of the fire PSA.

After this selection, 202 fire events remained in the database. For each fire, the following are documented :

- site and unit affected by the fire,
- date,
- reactor condition preceding the fire,
- location of fire (building),
- equipment or maintenance operation causing the fire.

2.2 Results

It has to be noted that the number of reported fire events were different for 900 MWe and 1300 MWe standardized plants. Concerning 900 MWe reactors, 186 fire events were reported for an operation period of 399 reactor-years whereas 93 fire events were declared in 109,2 reactor-years of operation for 1300 MWe reactors. The resulting fire frequencies are then 4,7 x 10^{-1} / reactor-year and 8,5 x 10^{-1} / reactor-year respectively, all reactor states included.

2.2.1 Fire cause

Table I - Fire cause distribution

FIRE CAUSE	NUMBER OF FIRE EVENTS	PERCENTAGE (%)
electrical	123	60,9
mechanichal	22	10,9
human intervention	46	22,7
hydrogen	6*	3
miscellaneous	5	2,5

* : fire events due to turbogenerator not included

Experience shows that electrical faults (short circuits, arcing, poor contacts) are the main cause of fires. The significant contribution of maintenance operations (grinding, cutting, welding, cleaning with solvents), which represent 22,7% of overall fire events, should also be noted.

Fires with mechanical origin are less frequent (around 11%); they result, to a large extent, from contact of flammable material with hot surfaces.

2.2.2 Fire location

Table II - Fire location distribution

FIRE PLANT LOCATION	NUMBER OF FIRE EVENTS	PERCENTAGE (%)
Nuclear Auxiliary Building	24	11,9
Fuel Building	25	12,4
Electrical Building	33	16,4
Reactor Building	32	15,8
Diesel Generator Building	9	4,4
Service Water Pumphouse	3	1,5
Turbine Building*	53	26,2
Others**	14	7
Location unknown	9	4,4

* : building outside the scope of the fire PSA.
** : Control Building, auxiliary, unit and main power transformers.

2.2.3 Reactor operating condition at the time of the fire

Table III - Reactor state distribution

REACTOR OPERATING CONDITION	NUMBER OF FIRE EVENTS	PERCENTAGE (%)
Refueling shutdown	54	26,7
Cold shutdown for maintenance	3	1,5
Normal cold shutdown	10	4,9
Intermediate shutdown, RRA connected	1	0,5
Intermediate shutdown, RRA conditions	2	1
Intermediate shutdown, RRA not connected	2	1
Hot shutdown	6	3
Hot standby	1	0,5
Reactor at power	103	51
Reactor operating condition unknown	20	9,9

Fires mainly occurred when the reactor was at power (51% of fires) or in a cold shutdown condition (33,1% of fires). This is due to the high proportion of time the reactors are at power and to the significant increase in the number of maintenance operations during cold shutdowns.

The distribution of fires that occurred in a cold shutdown state is the following :

- normal cold shutdown : 15%,
- cold shutdown for maintenance : 5%,
- refueling shutdown : 80%.

Equipment or maintenance operations initiating fires :

The analysis of fires made it possible, in most cases, to identify the equipment or maintenance operation that caused the fire. The various types are listed below.

1 Fire due to equipment fault

(a) Electrical equipment

Electrical energy conversion equipment:

- high and medium voltage equipment (6.6 kV controller, main transformer, step-down transformer, switchboard),
- turbogenerator including hydrogen leaks (electrical turning gear),
- diesel generators (oil leak),
- electrical switchboard (220 V switchboard, transformer, 380 V controller).

Instrumentation-control equipment:

- decentralized control console,
- instrumentation and control cabinet, regulating cabinet.

Actuators:

- motors (air compressor).

Miscellaneous:

- electric heaters, resistors.

(b) Pumps (primary pumps and turbine pumps of auxiliary feedwater system following an oil leak).

(c) Fans (fan motors or belts).

Nota : it must be noted that electric cables (48 V, 380 V, 6.6 kV) are not considered as fire initiators. The reason for this hypothesis is that the few fires which broke out on cables were due either to a mechanical shock which damaged the cable and caused a short-circuit or to the wrapping of a high voltage cable which heated up because of the Joule effect.

2 Fire due to hydrogen

Hydrogen leak (H_2 explosion in a tank)

3 Fire due to maintenance

Maintenance work (welding, grinding, solvent ignition in cleaning operations)

3. FIRE FREQUENCIES

3.1 Basic data

To facilitate the use of PWR operating experience and the evaluation of the frequencies of fires, equipment items of the same type were assigned to a reference group. This methodology allows to have for each ignition source the largest experience feedback and improves the reliability of the results.

For each reference group, a fire frequency is obtained from the operating experience with the following formula :

$$FRi = \frac{Nf}{T}$$

where Nf is the number of fire events related to the equipment of the reference group and T is the total operating time of french PWR's (508,2 reactor-years).

Examples of reference groups and the corresponding fire frequencies are given in the following table.

Table IV - Reference groups fire frequencies

REFERENCE GROUP	NUMBER OF FIRE EVENTS	FIRE FREQUENCY (/reactor-year)
Equipment:		
- High voltage equipment	9	1.8×10^{-2}
- Turbo generator	24	4.7×10^{-2}
- Diesel generator sets	6	1.2×10^{-2}
- Medium and low voltage equipment	28	5.5×10^{-2}
- Pumps	13	2.6×10^{-2}
- Electric heaters	13	2.6×10^{-2}
- Electric motors	5	9.8×10^{-3}
- Fans	17	3.3×10^{-2}
Maintenance:		
- Reactor at power	13	2.6×10^{-2}
- Reactor in shutdown states	31	6.1×10^{-2}

3.2 Fire frequency estimation method

Reactor at power

The estimation of the fire frequency in a room has to take into account two contributions : one due to equipment, the other due to maintenance operations.

Equipment

For each reference group, it is considered that the frequency of a fire due to the equipment is proportional to the number of equipment items located in the room.

For each reference group, the following data is then needed :

- the fire frequency of the reference group (FRi),
- the number of equipment items of the reference group in the room (Ni),
- the number of equipment items of the reference group in the plant (NTi).

The contribution of this reference group to the fire frequency in the room is then :

$$Fi = FRi \; x \; \frac{Ni}{NTi}$$

It must be noted that this method only concerns the buildings where a fire could threaten the safety of the plant. For a standard french 900 MWe PWR, it represents 822 rooms.

Knowing the contribution of each reference group i, the fire frequency in a room due to its equipment (F1) is obtained by summing the contributions Fi of the different groups :

$$F1 = \sum_i Fi$$

Maintenance operations

As the number of maintenance operations in a room is not known, it is considered, as a first approximation, that their contribution is uniform throughout the installation. If MP is the reference frequency, when the plant is at power, the contribution of maintenance operations to the fire frequency in a room is :

$$F1 = \frac{MP}{NL}$$

where NL is the total number of rooms in the installation where maintenance operations were performed. (NL = 664 for a standard french 900 MWe PWR, the rooms of the reactor building being excluded)

Frequency of a fire

The fire frequency in a room is then obtained by summing the contributions due to equipment and to maintenance operations :

$$FP = F1 + \frac{MP}{NL}$$

3.3 Results

The method for estimating the frequency of a fire was applied to unit 1 of the Blayais power station (900 MWe PWR) for all the rooms containing safety-related equipment. Implementing this method required to create an equipment database, allowing the location of each equipment item for the reactor. This database was established both from existing computer files (electrical cable listings and maintenance operation data used by the operator) and from data obtained by on-site visits. So it was possible to evaluate, for each reference group, the number of equipment items in each room and in all the plant. The approach was completely computerized using a database management software program which made it easy to calculate the frequency and to provide for possible updating.

For the Blayais station, with the reactor at power, the order of magnitude of fires frequencies obtained varies from 10^{-5} to 10^{-2}/reactor year, depending on the room content.

The distribution of rooms as a function of the different frequency ranges is shown in the following table :

Table V - Rooms distribution according to frequency range

FREQUENCY-RANGE (/reactor-year)	NUMBER OF ROOMS
$10^{-5} < f < 10^{-4}$	*166*
$10^{-4} < f < 10^{-3}$	*106*
$10^{-3} < f < 10^{-2}$	*40*

It must be noted that the total number of rooms is 312 which corresponds to the rooms where a fire could cause the loss of cables or equipment necessary to the reactor operation.

4. CONCLUSION

The analysis of fire-related operating experience provided information for carrying out the fire probability study. It is actually possible to identify the equipment items and maintenance operations which constitute the principal ignition sources for PWRs. It also allowed to attribute a fire frequency to each ignition source. These informations were used to develop the method for estimating the fire frequency for a room. This method allows to obtain reliable data due both to the homogeneous design of the French PWRs and to the considerable operating experience gained from more than 500 reactor-years. The application of this method to unit 1 of the Blayais power station shows that the approach developed by IPSN is easy to implement. The fire frequency for each room will be used to select the critical areas (rooms where the contribution of fires to the probability of core meltdown is not negligible).

Moreover, equipment or materials with an appreciable energy content, such as flammable fluids, near an ignition source constitute potential fires that should be studied as initiators of fire scenarios. The data acquired allow to estimate frequencies for these points and will be used to quantify the probabilities of fire scenarios liable to induce core meltdown.

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CRITERIA FOR CLASSIFICATION AND REPORTING OF FIRE INCIDENCES IN NUCLEAR POWER PLANTS OF INDIA

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Abstract

It is important that all fires in and around fire effective neighbourhood of Nuclear Power Plant (NPP) should be promptly reported (Reportable fires) and investigated. However, the depth of investigation and the range of authorities to whom the individual fire incidence need to be reported depends upon the severity of fire. In case of conventional non-chemical industries, the severity of fire depends mainly on the extent of loss caused by fire on property and the burn injury to persons. In case of NPP, two additional losses viz., release of radioactivity to working/public environment and the risk to safety related systems of NPP due to fire assume greater importance. This paper describes the criteria used in NPPs of India for classification of reportable fire incidences into four categories, viz., Insignificant, small, medium and large fires. It also gives the level of investigation depending upon the severity of fire. The fire classification scheme is explained in this paper with the help of worked out examples and two incidences of fire in Indian NPPs.

1. RELEVANCE OF RECORDS OF FIRE INCIDENCES AND ITS ANALYSIS

A systematic record of fire incidences is important for deciding the extent of upgradation needed for fire protection and also as an input data for carrying out Probablistic Safety Analysis of Nuclear Power Plant (NPP). Hence, it becomes essential that all fire incidences in and around fire effective neighbourhood of NPP's should be reported, recorded and investigated. Since fire destroys many of the evidences, estimating the root causes of fire generally becomes a time consuming activity. Further, in view of the impact a fire has on safety of NPP, the information on fire incidences is required to be given to different levels of management and regulators. This results in collection of large number of data (as minor fires of no significance also get reported) of fire incidences and investigation of these by different levels of management and regulators. Hence, a need has been felt to develop a criteria for classification of fire incidences, the levels of management up to which fire incidence should be reported, and to decide about the level of experts required to investigate the root causes of the fire incidence in NPPs.

2. CLASSIFICATION OF FIRE INCIDENCES

A criteria was developed to be used in Nuclear Power Plants (NPP) of India for classification of fire incidences and for its reporting. The aim in developing this criteria was that all fires in and around the fire effective neighbourhood of NPP should be reported and investigated. The level of investigation and reporting depends upon the severity of fire as defined by its class. Any other fire outside the above referred domain is called as non-reportable fire. Such non reportable fires are mainly due to burning of dry grass and are categorized as grass fires.

The reportable fires are classified into the following four categories based on the extent of damages and risks involved.

a) Insignificant fires

b) Small fires

c) Medium fires

d) Large fires

The following types of damages and importance of safety systems are considered for categorizing the fires in the above referred classes.

- Amount of financial loss
- Burn injury to persons
- Release of radioactivity to working/public environment
- Risk to the safety related systems.

The classification of fires based on type of loss/location area is given in the table I. Each fire taking place within the plant boundary and other plant related areas (such as stores, water intake system, waste management facilities etc.) except residential colony is analysed for identifying its class using this table.

TABLE I. CLASSIFICATION OF FIRES

Type of loss/Location			Class of fire			
			Insignificant	Small	Medium	Large
Financial loss due to fire			< Rs. 10,000/-	Between Rs. 10,000/- and Rs. 5 lakhs/-	Between Rs. 5.1 lakhs and Rs. 1 crore	> Rs. 1 crore
Extent of burn injury			No burn injury	Man-hour] < 8 lost due to] Man burn injury] days	Man-hour lost] > 8 due to burn] Man injury] days But not fatal	Burn injury leading to fatality
Release of activity due to fire	Plant working environment		Within normal permissible limit	Between 1 to 10 times the limit	Between 1 to 100 times the limit	> 100 times the limit
	Public environment .		Within normal permissible limit	Between 1 to 2 times the limit	Between 2 to 10 times the limit	> 10 times the limit
LOCATION	Non-Safety related area		Fire got extinguished by itself or by using portable fire extinguisher.	Fire got extinguished only after use of fixed suppression system or fire tender.		
	Safety related areas	Safety equipment Not involved in fire	Fire got extinguished by itself.	Fire got extinguished by using portable fire extinguishers	Fire got extinguished by using fixed suppression system or by using fire tender.	
		Safety equipment involved in fire		Fire got extinguished by itself.	Fire got extinguished by using portable fire extinguisher.	Fire got extinguished by using fixed suppression system or by using fire tender.

- Note : 1. Rs. Rupee, an Indian currency
  2. 1 Lakh = 0.1 million
  3. 1 Crore = 10 million

TABLE II-a. EXAMPLES FOR CLASSIFICATION OF FIRE
DATA ON FIRE

Ex. No.	Financial loss (Rupees)	Burn Injury (Man days lost)	Activity released (limit)		Location			Extinguishing Style
			Plant	Public	Safety related area (Safety Equipment)		Non Safety related area	
					involved	not involved		
1	Rs. 1 lakh	7 days	5 times	< limit			Yes	portable extinguisher
2	Rs. 2 crores	No burn injury	NIL	NIL			Yes	fire hydrant
3	NIL	Fatal	NIL	NIL			Yes	extinguished itself
4	NIL	No burn injury	2000 times	< limit			Yes	portable extinguisher
5	NIL	No burn injury	NIL	NIL		Yes		extinguished itself
6	NIL	No burn injury	NIL	NIL	Yes			extinguished itself
7	NIL	No burn injury	NIL	NIL	Yes			portable extinguisher
8	NIL	No burn injury	NIL	NIL	Yes			fire tender
9	Rs. 5 Crores	Fatal	5000 times	100 times	Yes			fire tender
10	Rs. 2 lakh	No burn injury	1000 times	3 times	Yes			fire tender
11	NIL	No burn injury	20 times	NIL	Yes			portable extinguisher
12	10 lakhs	100 days	40 times	NIL		Yes		fixed fire suppression system
13	50 lakhs	20 days	100 times	15 times	Yes			fire tender
14	80 lakhs	200 days	15 times	1.5 times	Yes			fire tender
15	10 lakhs	25 days	NIL	NIL		Yes		fire tender
16	8 lakhs	Fatal	NIL	NIL			Yes	fire tender
17	NIL	No burn injury	12 times	2 times			Yes	fire tender
18	1 lakh	Fatal	NIL	NIL			Yes	fire tender

1 US Dollar = 37 Rupees　　Lakh = 0.1 million　　Crore = 10 million

TABLE II-b. EXAMPLES FOR CLASSIFICATION OF FIRE
ASSESSED CLASS OF FIRE

Ex. NO.	Finance	Burn Injury	Activity		Location	Final class of fire	Remarks
			Plant	Public			
1	Small	Small	Small	Insignificant	Insignificant	Small	
2	Large	Insignificant	Insignificant	Insignificant	Small	Large	
3	Insignificant	Large	Insignificant	Insignificant	Insignificant	Large	
4	Insignificant	Insignificant	Large	Insignificant	Insignificant	Large	
5	Insignificant	Insignificant	Insignificant	Insignificant	Insignificant	Insignificant	
6	Insignificant	Insignificant	Insignificant	Insignificant	Small	Small	
7	Insignificant	Insignificant	Insignificant	Insignificant	Medium	Medium	
8	Insignificant	Insignificant	Insignificant	Insignificant	Large	Large	
9	Large	Large	Large	Large	Large	Large	
10	Small	Insignificant	Large	Medium	Large	Large	
11	Insignificant	Insignificant	Medium	Insignificant	Medium	Medium	
12	Medium	Medium	Medium	Insignificant	Medium	Medium	
13	Medium	Medium	Medium	Large	Large	Large	
14	Medium	Medium	Medium	Small	Large	Large	
15	Medium	Medium	Insignificant	Insignificant	Large	Large	
16	Medium	Large	Insignificant	Insignificant	Small	Large	Grass fire in store yard
17	Insignificant	Insignificant	Medium	Small	Small	Medium	Grass fire in waste burial ground
18	Small	Large	Insignificant	Insignificant	Small	Non reportable	**

** Grass fire in colony contract labour died. The worth destroyed is Rs.1 lakh, this is classified as non-reportable because it is away from plant premises

3. ASSIGNING 'CLASS' TO FIRE INCIDENCES

The classification of fire is to be done based on all the types of damages/location using Table I. Which ever assessment gives the highest severity of fire, the fire incidence in analysis is assigned that severe class. To facilitate users for assigning 'class' of fire to fire incidences, illustrative (hypothetical) cases of fire are explained by worked out examples in Tables II - a and II- b. The data on these fire incidences is given in Table II-a and assessment of 'class' for the incidence based on the extent of financial loss/burn injury/activity release/location is given in table II -b. It can be seen from these tables that if any fire incidence involved safety related equipment, the fire incidence falls in higher severity class. A typical example is no.8, which gave a class as 'Insignificant' based on financial loss, burn injury, activity release but classed as 'Large' since the fire involved safety related equipment and could only be extinguished using fire tender. Similarly, higher class is assigned to fire if activity is released (example no.10). A typical case of grass fire is given in example No.16 wherein fire has been assigned a class 'Large' as it resulted in a fatality. Whereas in another case (example no.17) of grass fire, though no financial loss/no burn injury/non involvement of safety related equipment occurred, but due to activity release it is assigned higher class.

4. REPORTING AND ANALYSIS OF FIRE INCIDENCES

The reporting and analysis criteria of fires is given in Table III . There is a local plant fire safety committee at each of the NPP, which investigates all the insignificant and small fires including grass fires and these are not reported to Corporate office and to the regulatory authorities. Special fire investigation committee consisting of members from that NPP are instituted for investigating each of the medium fires. Members from units other than the plant are also included in the special fire investigation committee instituted for investigation of each of the large fires. The fire incidences particularly falling in

TABLE III. REPORTING AND ANALYSIS CRITERIA OF FIRES

Nature of action	Grass fires	Insignificant	Small	Medium	Large
Reporting to	Local plant Managemen t	Local Plant Management	Local Plant Management	Local Plant Management Corporate office AERB DAE Apex Fire Safety Committee	Local Plant Management Corporate office AERB DAE Apex Fire Safety Committee Concerned statutory body.
Investigation by	Local fire Safety Committee	Local fire Safety Committee	Local fire Safety Committee	Special investigation committee consisting of plant members.	Special investigation committee consisting of members also from units other than the plant.

Note : - 1. AERB - Atomic Energy Regulatory Board

2. DAE - Department of Atomic Energy

medium and large fire categories are also investigated by committee instituted by Atomic Energy Regulatory Board (AERB). Another higher level committee called Department of Atomic Energy's (DAE) Apex Fire Safety Committee periodically discusses the fire incidences and makes appropriate recommendations applicable to the NPPs and to other plants of DAE to prevent occurrence of similar fire incidences. Each of the medium and large fires are promptly (within 24 hours) reported to Corporate office and to the concerned statutory authorities.

5. ANALYSIS OF SOME FIRE INCIDENCES IN INDIA'S NAPP'S

The criteria described in this paper has been put in practise at NPPs of India in 1997, and no significant fire has occurred as yet during this year. However, two incidences of fire which occurred in earlier years are described in this section. As yet no fires have occurred in India's NPPs involving burn injuries or release of radioactivity, still some fires falling in class of 'Large' have occurred mainly because it affected the safety related system and resulted in financial losses.

5.1 *Cable fire at RAPS-2*

Rajasthan Atomic Power Station Unit-2 (RAPS -2), a 220 Mwe Nuclear Power Station is located at Rajasthan, India. The incident occurred on 25th July, 1985, when the Station was operating at 200 Mwe. One primary circulating pump tripped on instantaneous over current. This was followed by a reactor trip. Several annunciations came in simultaneously. Attempt was made to restart the pump, (on hindsight this is conjured as a wrong step). Three more pumps tripped afterwards. Fast cool down of the reactor was initiated. Entry was made into Boiler room inside Reactor Building. Thick smoke resulted in extremely poor visibility. The emergency core injection alarm came in spuriously. Due to this, the shutdown cooling pumps had to be started after jumping the logic. During design stage no elegant arrangement was made for purging the smoke and entry without BA set was possible only after about 7 hours. Core cooling was adequately taken care of and no radiation spread was there. Unit was restarted after 73 days of repair/modification work.

The classification of this fire incidence of RAPS as per different types of damages is as follows:

Financial loss - Large fire

Burn Injury - Insignificant fire

Activity release - Insignificant fire

Safety related equipment - Large fire

Final class - Large fire

5.2 *Narora fire incident*

The Narora Atomic Power Station (NAPS) is a twin - unit, (220 Mwe each) PHWR located in Uttar Pradesh, India. Unit 1&2 of NAPS were commissioned in the year 1989 and 1992 respectively. On March 31, 1993, a fire occurred in the turbine building of NAPS-1. Failure of turbine blades in the 5th stage of flow path-2 of the LP turbine was identified as the cause of the incident. This led to failure of the generator hydrogen seals and the escaping hydrogen got ignited resulting in a large fire. The fire spread to several power and control cables in the turbine building and caused complete loss of power supply which lasted for a period of 17 hours. The reactor was immediately tripped and could be maintained in a safe shut-down state with adequate sub-criticality at all stages subsequently. Crash cool down of the primary heat transport system was done and subsequently fire fighting water was injected into the steam generators for maintaining thermosyphon cooling of the core. The fire was confined within the turbine building. Control power supply cable trays on the mezzanine floor (106M elevation) got severely affected by fire. Turbo-generator (TG) support structure and portion of the slab around TG set suffered damage due to intense heat of fire. Number of window glass panes in turbine building were found shattered.

Cable fire, lack of proper fire-retarding provisions and inadequacy in fire-barriers together with insufficient physical separation in redundant safety related cables was identified as the main cause of the extended station black out and consequent degradation of several safety systems. However, there was no radiological impact due to the incident, either on the plant or in the public domain and there was no loss of life or injury to personnel. The incident was categorised at level-3 on the International Nuclear Event Scale.

The classification of this fire incidence of NAPS as per different types of damages is as follows:

Financial loss	-	Large fire
Burn injury	-	Insignificant fire
Activity release	-	Insignificant fire
Safety related equipment	-	Large fire
Final class	-	Large fire

5.3 *Reporting and investigation of RAPS and NAPS fire incidences*

Though the present criteria of classification and reporting of fire incidences was not in practice at the time of these two fire incidences, these were reported and investigated as per the criteria given in this paper. The fires were treated in 'Large' fires class, reported immediately to concerned management and regulator. Special committees were instituted to investigate these incidences.

6. CONCLUDING REMARKS

The criteria given in this paper is basically classifying fires into a very limited number of categories. On one hand the classification scheme should be simple, at the same time one should be able to clearly distinguish between fires of different severity. The last requirement may necessitate even a finer classification among 'Large' fires. For example, the Narora fire case given in this paper is comparatively of higher severity than that of RAPS cable fire. In case, the fire incidence results in burn injury including fatalities along with large scale release of radioactivity (as has been the case with chernobyl), the severity is infact very high. Apparently, there is a need to further develop the fire classification criteria and better would be to have an internationally accepted standard on similar lines as International Nuclear Event Scale (INES). This could be considered by IAEA for developing a guide on this subject.

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RISKS OF TURBINE GENERATORS AT VVER-440 NUCLEAR POWER PLANTS

XA9847519

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Abstract

Many serious fires and incidents have occurred in the turbine halls of nuclear power plants, resulting in serious damage and long shutdown outages Some of these incidents have endangered the safe shutdown of the plants because of the location of lack of vital fire protection safety systems A detailed analysis is necessary for all those plants that have equipment important for safe shutdown located in the turbine hall or its vicinity without strict fire separation by fire rated barriers A reduction in the fire frequencies of the turbine hall is an additional way of improving safety This is possible by improving all aspects of turbine generator operation

1 INTRODUCTION

It was earlier generally assumed that turbine generators are part of an NPP's conventional systems which are not important to nuclear safety However, operational experience has indicated that disturbances of turbine generator systems often affect overall plant safety Many disturbances have led to accidents causing extensive damage in turbine halls and in their vicinity The worst consequences have been loss of habitability of the control room, loss of residual heat removal or total loss of electrical power for a longer period

Operational experience has also shown, that frequencies for events initiating from turbine generators are much higher than estimated in the design phase The most serious consequences of turbine generator failures seem to be fires, which also have a very high event frequency Some disturbances and accidents have taken place in Soviet-designed plants In VVER-440-plants, turbine generator damage can lead to the loss of the main feed water, emergency feed water and primary circuit residual heat removal functions because these systems are located in the turbine hall

This paper is based on extensive research, the goal of which has been to assess the operational safety of turbine generators at VVER-440 plants, especially at Loviisa NPP In this paper, the operation, monitoring testing and inspections of the most significant turbine generator systems have been studied Taking these findings into consideration and by using operational data from Loviisa and other power plants, the most significant safety issues of the turbine generator systems have been identified The frequency of initiating events and their possible consequences have been determined by using operational experience data and relevant literature

2 VVER-440 TURBINE HALL FEATURES

2.1 Lay-out considerations

The original VVER-440 PWR plant concept consists typically of two reactors sharing a common turbine hall Steam generated by the six steam generators of the one reactor streams to two turbine generators, which are located axially to the control (or intermediate) building The turbine building is an open hall without any walls or fire barriers between the turbine generators The control rooms, relay and switchboard rooms as well as cable rooms are situated inside the same building complex, in a so-called intermediate (or deaerator) building

The load bearing structures and roof beams of the integrated turbine hall and intermediate building are made of uninsulated steel constructions Thus, severe damage or even collapsing of the structures or roof in a big turbine hall fire is possible in less than ten minutes, as fire analyses as well as accidents occurred at similarly constructed Soviet-designed NPPs prove

Many important components are located in the open space of the turbine hall e.g. the main and auxiliary feed water pumps, pipelines and valves, power and control cables of these systems as well as condensate pumps Furthermore, many cable raceways that serve other safety-related systems are located in the open space of the turbine hall According to the original design, none of these systems is protected against fires or collapsing structures Fire may also spread to other compartments through inadequately sealed penetrations along the cable raceways as well as inadequate, open or missing fire doors. In addition, smoke may spread to the intermediate building and possibly jeopardize the habitability of the control rooms

Loviisa NPP is modified and improved from the original VVER-440 plant concept (see the paper "Safety improvements at Loviisa NPP to reduce fire risks originating from turbine generators"). The Loviisa reactors are situated in independent reactor containment buildings and the turbine generators are located in a favourable position transversally to the reactor buildings thus reducing the danger of missile ejection from turbine Fire protection has been accounted for already in the original design and further significantly improved during the plant's operation

2.2 Description of turbine generator systems

This chapter describes mainly the turbine generator systems as constructed at Loviisa NPP Some systems differ substantially from other VVER-440 plants e g hydrogen supply piping, fire valve in the governing oil line etc The turbine generators are also protected with local and area sprinkler systems

The Loviisa turbines comprise of one high pressure (HP) turbine case and two low pressure (LP) cases After the HP turbine there are two superheaters next to each other The generator and turbine are interconnected by a common shaft Their speed of rotation is 3000 1/min

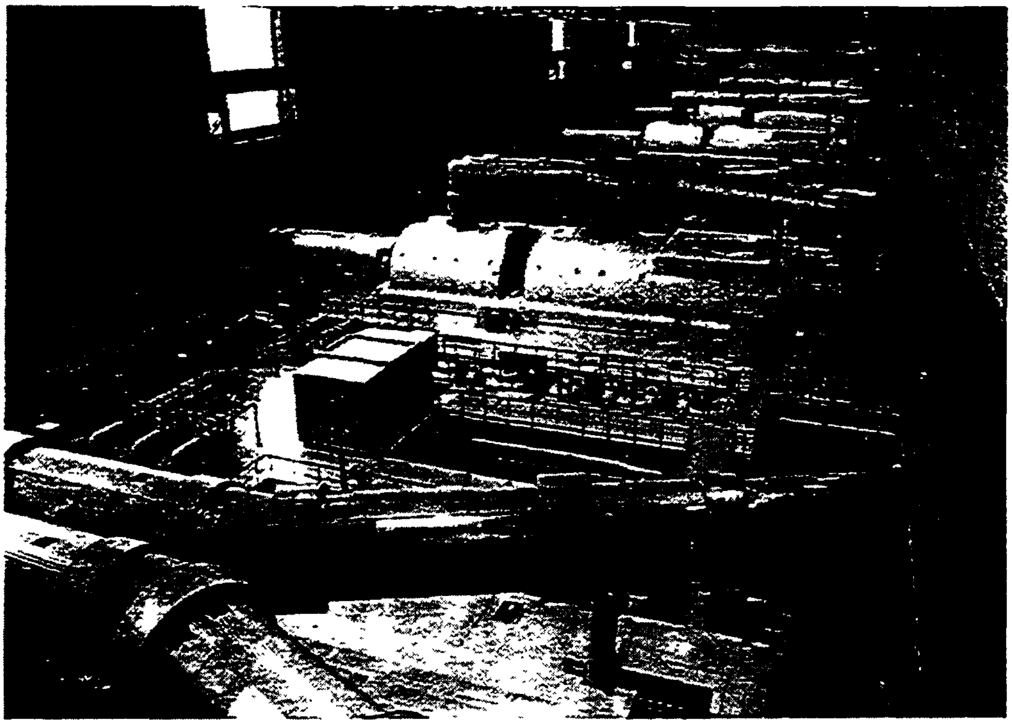
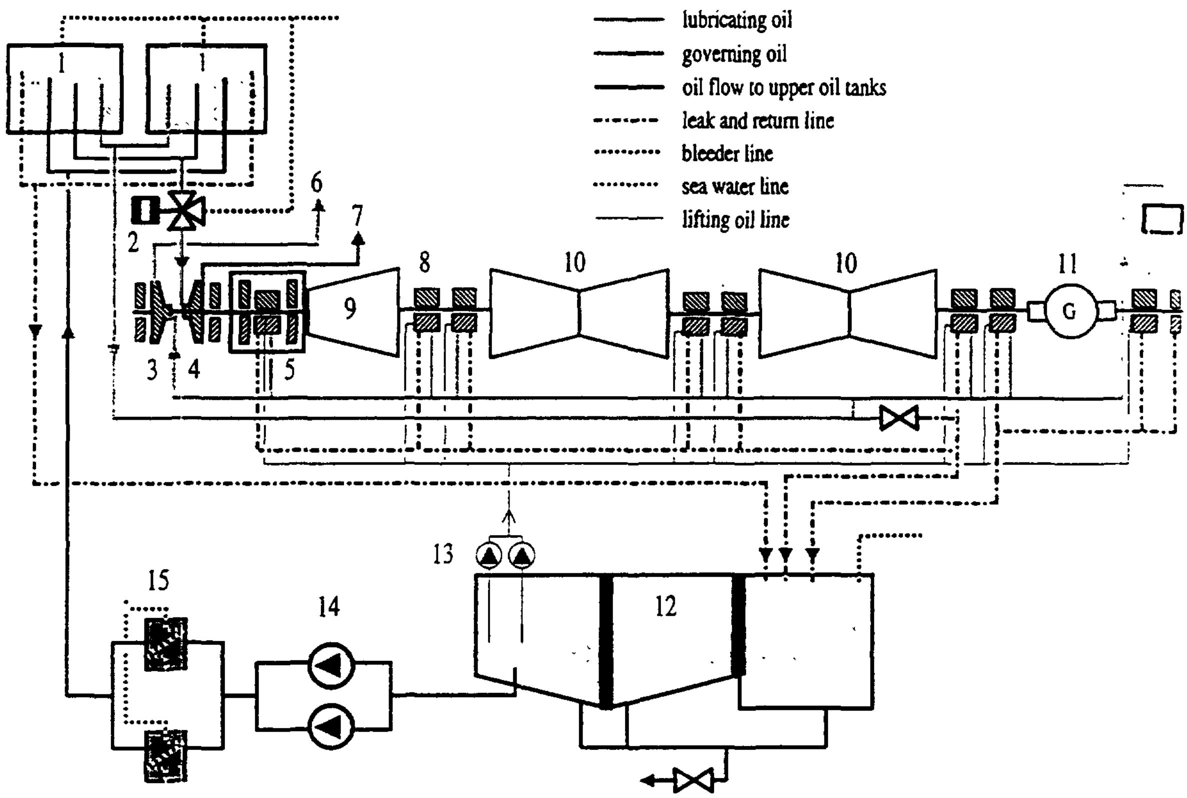


Figure 1. *Loviisa turbine hall.*

The auxiliary of the turbine generators systems are:

- **turbine:**
  - lubricating oil system (includes also lifting oil system)
  - governing and protection oil system
  - condensing system and ejectors
  - gland steam system
  - turbine drain system
- **generator:**
  - excitation system
  - stator water cooling system
  - hydrogen cooling system
  - sealing oil system.

Considering fire risks, the most risk-significant auxiliary systems are oil systems and the hydrogen cooling system. Lubricating oil is a big fire load ($56m^3$ per turbine generator, corresponding to a thermal power of 190 MW with burn-out of three hours). Hydrogen's fire load is minor, but due to its sensitivity for explosion the risks are also significant. Other notable fire loads are the lubricating oil of the pumps (for instance the main feed water pumps) and the electrical components (cables, motors etc.) which are located everywhere in the turbine hall.



1	*upper oil tanks 2 · 8,5 m³*	*9*	*high pressure turbine*
2	*fire valve*	*10*	*low pressure turbines*
3	*impeller pump*	*11*	*generator*
4	*governing oil pump*	*12*	*main oil tank 56 m³*
5	*integrated axial and radial bearings*	*13*	*lifting oil pumps 2 · 100 %*
6	*signal oil to hydraulic speed governor*	*14*	*main oil pumps 2 · 100 %*
7	*governing and protection oil feeding*	*15*	*oil coolers 2 · 100 %*
8	*radial bearing*		

Figure 2. *Lubricating system.*

2.2.1 Lubricating systems

Lubricating oil to a turbine generator's bearings flows from two upper oil tanks (gravity tanks), which are situated on the turbine hall's inside wall. Lubricating oil is pumped from the main oil tank into the upper oil tanks by main oil pumps (fig. 2). The main oil tank is located beside the generator, but on a lower level. The upper oil tanks are dimensioned so that if the oil flow from the main oil pumps stops, bearings lubrication can be maintained during turbine generator shut-down.

In shut-down or start-up phase, normal lubrication is insufficient due to low rotation speed. Therefore the lifting oil system feeds oil to the lower coating of the bearings with a 5,0 MPa pressure for maintaining an oil film between bearing and shaft babbitt metal surfaces.

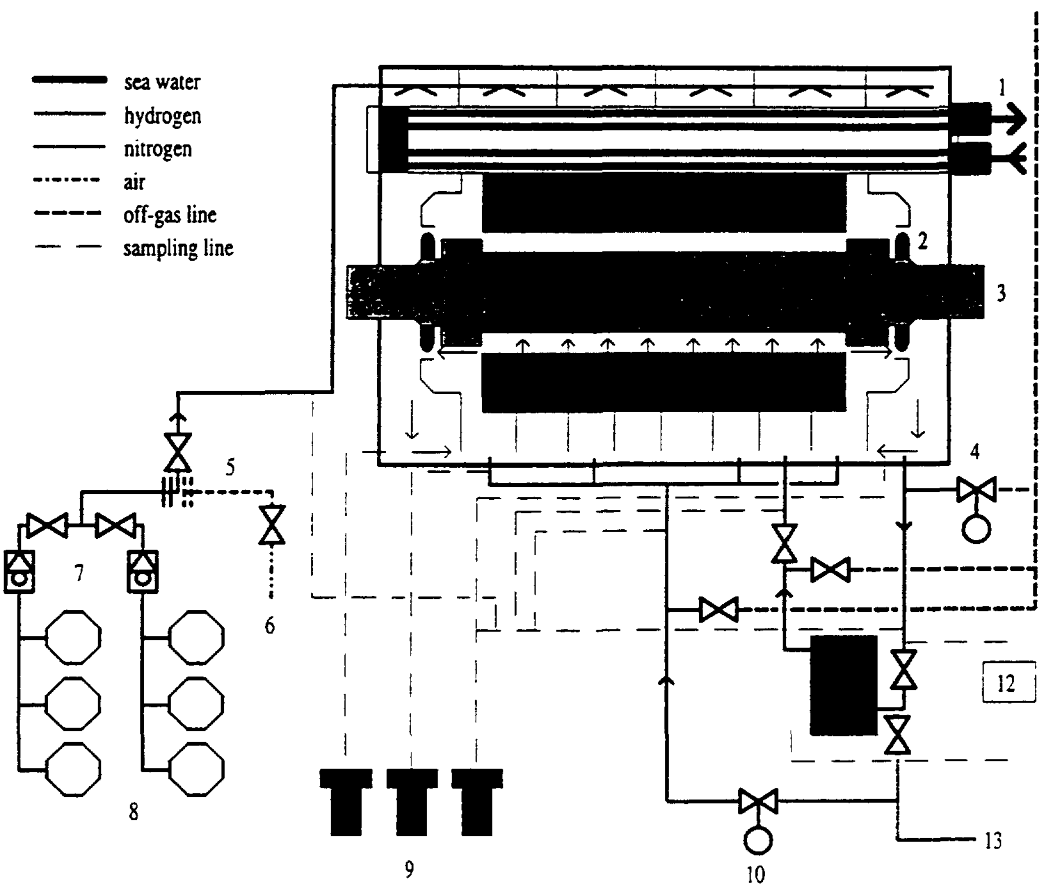
2.2.2 Generator sealing oil and hydrogen cooling systems

The Generator rotor is cooled by hydrogen. Hydrogen, the pressure of which is about 0,4 MPa and volume about $225 m^3_{NTP}$, circulates inside the generator through the air gap and the stator's table packs. The flow is induced by two blowers, which are situated at the end of the generator's shaft (fig. 3). The hydrogen is cooled by four coolers on the generator's top. The generator's shaft penetrations are sealed by sealing oil to prevent hydrogen leaks from the generator to the turbine hall or the exciter cover.

2.2.3 *Turbine generator's protection systems*

A turbine trip is triggered by the following signals:

- **protection signals from process**
    - insufficient vacuum in the condenser
    - high water level in superheater collectors
    - low pressure in the governing oil system (the pressure decreases also when the fire valve actuates (fig. 2) due to a fire in HP turbine)



1	*hydrogen coolers*	*8*	*hydrogen bottles*
2	*blowers*	*9*	*liquid detectors*
3	*rotor*	*10*	*emergency nitrogen valve*
4	*valve for hydrogen discharge*	*11*	*hydrogen dryer*
5	*mechanical difference*	*12*	*purity analyser*
6	*air feed*	*13*	*nitrogen feed*
7	*automatic cut-off valves*		

Figure 3. *Hydrogen cooling system.*

- low steam pressure in HP turbine inlet
- high pressure in HP turbine regulating chamber
- high steam pressure in HP turbine outlet
- high water level in steam generators
- power difference between primary and secondary circuits

- **other protection signals**
  - axial movement of turbine generator shaft
  - operation of turbine's mechanical overspeed device
  - failure of hydraulic speed governor
  - closure of superheater's butterfly valves during power operation
  - closure of at least two out of four stop valves
  - manual trip from control room
  - manual trip locally from turbine
  - reactor shutdown
- **signal from generator's protections**
- **signal from transformers' protections.**

The majority of the generator protections will also cause a turbine trip signal In a loss of off-site power situation the aim is that the unit would remains in the house-load operation mode Then the 400 kV main circuit breaker will open due to the operation of generator protections but turbine protection will not get a signal for turbine trip

3 MONITORING, TESTING AND INSPECTIONS OF TURBINE GENERATORS

3.1 Monitoring during power operation

The major monitoring objects for operational safety are

- temperature of turbine generator bearing babbitt metal and lubricating oil
- vibrations of rotor and bearings, especially rapid changes in vibration levels
- anomalous values and rapid changes of steam pressure and power output
- cleanliness of generator's carbon brush
- leak tightness of hydrogen and oil systems, the amount and location of the leak
- purity and humidity of hydrogen
- oil pressure in sealing oil system
- liquid leaks into generator

Many of the measurements mentioned above can be performed by using plain temperature or pressure sensors More advanced systems have to be utilized when vibration levels, purity or humidity values and gas or liquid leaks are rated

3.1.1 *Vibration monitoring*

Vibration monitoring observes the condition of turbine generators extensively. The Loviisa NPP units have some differences in monitoring systems. Loviisa-1 bearing vibrations are monitored by speed probes, which measure horizontal vibrations from all bearings and axial vibration from one bearing. Shaft vibrations are monitored by eddy current probes. They give both horizontal and vertical vibration levels which are relatively compared to bearing vibrations. The overall level of bearing vibrations is presented on the process computer display. The time delay between monitoring and display is rather long, about 20 minutes. In-situ values can only be read from the monitoring system's central unit in the back of the control room. Values from the probe are also led through the central unit to a condition monitoring system, which gives a report after 20 minutes' delay.

Loviisa-2 vibration monitoring system sensors are more advanced than those in Loviisa-1. Bearing vibrations are measured by accelerometers for all bearings in vertical direction and for two bearings in axial direction. Shaft vibration sensors measure relative values from both sides of turbine cases and generator in horizontal and vertical directions. Some improvements have been planned for vibration monitoring systems, especially at Loviisa-1. The length of time delays, the small quantity of the measuring sensors and the lack of an automatic turbine trip due to high vibration levels are the most severe drawbacks of the current-day vibration monitoring system.

3.2 Testing

Testing is usually done before or after annual maintenance outage. A few tests are carried out also during power operation (Table I). During shut-down or start-up the most important objects for testing are:

- action of the turbine generator protection system
- action of the mechanical overspeed device
- tightness and operation time of stop and regulating valves
- action of the hydraulic speed governing system
- action of turbine electrical-hydraulic governing system
- generator's tightness
- operability of generator emergency nitrogen feeding.

Table I. *Tests during power operation at Loviisa NPP.*

test	interval [weeks]
movement of stop valves	1
action of mechanical overspeed device	4
vibration monitoring by portable equipment	4

3.3 Surveillance inspections during annual maintenance outage

The most important objects of surveillance inspections are (testing intervals in brackets)

- HP turbine (5 years)
- LP turbines (10 years)
- generator (3 years)
- turbine generator bearings (2 years)
- generator seal bearings (1 year)
- turbine valves (2 years)
- components of the governing and the protection systems (2 years)
- centrifugal clutch of mechanical overspeed device (6 years)

The integrity of turbine blades is inspected every second year by endoscope through penetrations without opening the cases Because endoscope checking is quickly accomplished also rather short outages can be utilized A good example of the successful use of endoscope checking is the discovery of LP turbine blade cracking at Loviisa NPP (fig 4)

Figure 4. *Cracked LP turbine blade on endoscope display at Loviisa NPP.*

4 TYPICAL TURBINE GENERATOR ACCIDENTS

By studying turbine generator accidents at NPPs some typical root causes, failure modes and consequences can be found out. Some examples of these accidents are presented in Appendix I.

The identified turbine failure modes, which have led to serious accidents are:

- breaking of blades and bearings
- leaks from oil systems.

The breaking of turbine blades can be a consequence of turbine overspeed or the blades can crack due to shock action, fabrication defects or fatigue. The most hazardous part is the last zone of a LP turbine where the blades are long and the steam is wet. Breaking may cause:

- missile ejection from turbine ⇒ component damage, fires, floods
- loose parts hitting the condenser ⇒ loss of condenser
- unbalance of rotor ⇒ (seal) bearings breaking ⇒ hydrogen and oil leaks ⇒ ignition due to friction of heated bearing metal.

In governing oil system leaks the amount of leaked oil is small, but ignition is rather probable because of high oil pressure and the vicinity of hot surfaces (HP turbine, steam pipes and valves). In lubricating oil leaks oil flow can be very large, but immediate ignition is not so self-evident. On the other hand, oil soaked insulation material can self-ignite in lower temperatures. Prime causes for an oil leak may be:

- pipe vibrations due to turbine rotor vibrations
- insufficient or wrong bracing of pipes
- turbine missiles
- hydrogen explosion.

At some VVER-440 plants, phosphate ester based lubricants are used instead of mineral oils to reduce fire risks. Compared to mineral oils their flashpoint is somewhat higher but the fire point and autoignition temperatures are remarkably higher. The ignition of these fire resistant lubricants is possible in the case of bearing damage resulting to hot metal surfaces. The heat release rate will be about half of that of mineral oils.

Generator failure modes and consequences are:

- breaking of seal bearings ⇒ hydrogen and oil leaks ⇒ fires and explosions
- leakage of the sealing oil system ⇒ hydrogen and oil leaks ⇒ fires and explosions
- hydrogen cooling system damage ⇒ hydrogen leaks ⇒ fires and explosions
- rotor's breaking ⇒ hydrogen and oil leaks, missiles ⇒ fires, explosions and component damage
- fouling or damaging of the carbon brush mechanism ⇒ carbon brush fires and component damage.

Some rotor failures, which have been the consequence of a generator's incorrect connection to the electrical grid (Appendix I) have occurred to generators made in the former Soviet Union In the case of an incorrect connection the generator begins to work like an electric motor and picks up speed to the rated speed in tens of seconds Due to rapid acceleration enormous electrical and mechanical forces will be exerted to the rotor, which can lead to rotor unbalance The result has usually been serious generator damage and a following fire

5 EVENT FREQUENCIES OF TURBINE GENERATOR FAILURES

5.1 Event frequency of turbine missile accident

Traditionally, turbine generator systems risks have been assumed to originate from turbine missiles The primary concern has been the situation where missiles ejected from turbine hit the reactor building or other places significant for safety and make safety systems inoperable A research issued in the early 1970's [1] brought out, that the event frequency for a turbine missile accident can be calculated by the equation (1)

$$P_4 = P_1 \; P_2 \; P_3 \qquad (1)$$

where P_1 is frequency at which turbine missile will penetrate its casing

P_2 is probability that the missile will strike safety-related equipment

P_3 is probability that the strike will make safety-related equipment inoperable

P_4 is frequency of missiles making safety-related equipment inoperable

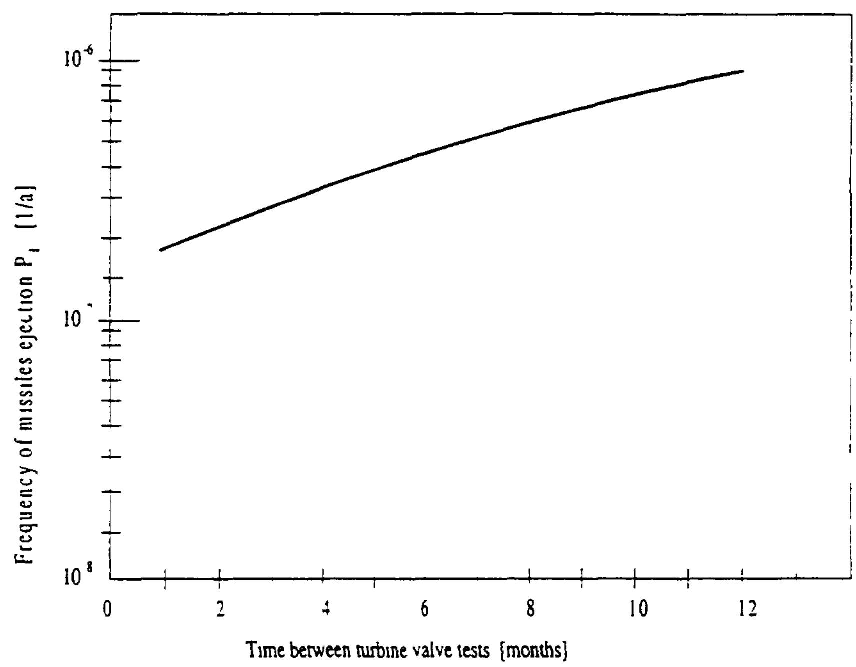


Figure 5. Estimate made by manufacturer for missile ejection frequency as a function of turbine valve test interval [2]

As is recognised in reference [1], the frequency of missile ejection (P_1) depends on the design of turbine rotor, speed of rotation and thickness of turbine casing. Conservative estimate for P_1 is 10^{-4} events per year. An equally conservative estimate for P_2 is 10^{-3}. It is also conservative to estimate that a missile breaks its target. If a turbine is oriented unfavourably towards safety-related systems, the frequency P_2 may be 10^{-1}. When referred to the above estimates, the frequency for missiles making safety-related equipment inoperable (P_4) will be 10^{-7} per year for favourably oriented turbines and a maximum of 10^{-5} for unfavourably oriented turbines.

The missile ejection frequency (P_1), based on operational experience, has a value as high as $1{,}25 \cdot 10^{-3}$ per year with a 90 percent confidence. The estimate has been gained by using operational data from Salem-2 and other similar turbines manufactured by Westinghouse. Therefore the result is valid only for identical turbines. According to [2] the ejection frequency P_1, being between 10^{-9} - 10^{-1}, depends not only on the turbine's structure and speed of rotation, but also on operating practices, tests and inspections. How the missile production is estimated to depend on the frequency of turbine valve tests at Salem NPP is brought out in figure 5. The evaluation has been made before the accident at Salem-2 (see Appendix I).

In [3] the effect of turbine (low-pressure) casing inspection intervals on the missile ejection frequency P_1 is recognised. By increasing the inspection period from 4,5 to 9 years, P_1 will quadruple. Inspections every 30 years increase the frequency of missile ejection over tenfold compared to a 4,5 years' checking interval.

Before the event the missile frequency of an individual plant or plant type can be estimated, the operational history, monitoring, protection, tests and inspections of turbine generators must be studied in addition to turbine generator plant features. Missile accidents or other rotational parts breaks have been rather unusual at VVER-440 plants. Therefore the initial value for the missile ejection frequency P_1 can be estimated to be 10^{-4} per year.

Hitting probability P_2 at Loviisa-1 is examined in [4] and [5]. Studied targets located in turbine hall are pumps and heat exchangers of the residual heat removal system and the conventional intermediate cooling system's upper water tanks. Other targets, beside the turbine hall, are service water system's pumps and the control room of the other unit, which are also in the range of the missile ejection angle. The order of magnitude of the probability is from 10^{-4} to 10^{-3}.

Referring to the information above, the frequency of the missile accident (P_4) is low, a conservative assumption is 10^{-7} per year at Loviisa NPP.

At plants, where the turbines are situated unfavourably (like at an original VVER-440 plant), turbine missile risks are higher due to the fact that more safety related equipment are located along the trajectory of the missiles, thus the probability P_2 is higher. Also, in the worst case scenario the heaviest missiles having an initial velocity of several hundreds m/s may penetrate tens of centimeters of concrete constructions, thus possibly striking also vital parts of the control building.

It is also important to notice the significance of operational procedures, tests, checks and monitoring By performing these functions in an appropriate manner the missile accident risk may be decreased by several decades

5.2 Turbine generator fires

5 2 1 Event frequency

Fire frequencies of turbine generator systems at American NPPs are calculated in [6] Event frequencies are based on 22017 years of operational experience from conventional power plants and on 577 years operational experience from nuclear power plants. The results are shown in Table II. As the frequencies show, the fire risk is higher for large turbine generators than for smaller ones. High NPP fire values are explained partly by large NPP turbine generators

Table II. *Event frequency of turbine generator fires [fires/year] [6]*

period	conventional plants [MW]				NPPs in all
	60 - 249	250 - 499	500 -	in all	
1940-1949	1,1 10^{2}	-	-	1,1 10^{2}	-
1950-1959	3,3 10^{3}	2,5 10^{2}	-	3,6 10^{3}	-
1960-1969	1,9 10^{3}	3,3 10^{3}	2,4 10^{2}	2,5 10^{3}	0
1970-1979	3,4 10^{3}	5,1 10^{3}	8,4 10^{3}	4,3 10^{3}	3 2 10^{2}
1980-1983	5,4 10^{3}	1,6 10^{2}	1,7 10^{2}	9,7 10^{3}	2,2 10^{2}
1940-1983	3,4 10^{3}	8,0 10^{3}	12,7 10^{3}	4,8 10^{3}	2,6 10^{2}

The fire frequencies of various buildings at American NPPs have been presented in [7] from year 1994 (Table III) Fire in the turbine hall seems to be the most common The turbine building fire frequency is higher in Table III than in Table II One reason is that the value in Table III includes also turbine building other fire than turbine generator system fires

Table III. *Fire frequencies at American NPPs.*

area	fire frequencies [fires / year]
turbine building	1,1 10^{-1}
auxiliary reactor building	7,0 10^{-2}
diesel generator room	2,6 10^{-2}
reactor building	1,7 10^{2}
control room	7,2 10^{3}
cable spreading room	4,3 10^{3}
service water building	2,0 10^{3}

According to insurance company statistics, the frequency of turbine generator system fires is 7,0 10^{-3} per year The frequency is calculated with data obtained by insurers (damages have exceeded the deductible) In the case of economically serious fires (more than two million dollars reclamation) the frequency has a value of 2,3 10^{3} per year

5 2 2 *Distribution of combustible material*

The most significant fires in [6] which have caused outages exceeding one day or economical losses of more than 100000 dollars are collected in Table IV According to [6] a fire has been significant in a fourth part of the cases The frequencies of different material fires can be estimated by accenting the turbine building fire frequency presented in Table III with the distribution of combustible material It is also assumed that the fire has been significant in a fourth part of cases

Table IV. *Distribution and fire frequency of combustible material at significant fires [6]*

combustible material	percentage [%]	frequency [fires/year]
oil	72,7	2,1 10^{-2}
hydrogen and oil	9,1	2,6 10^{3}
electrical components	9 1	2,6 10^{3}
hydrogen	7,3	2,1 10^{3}
wood	1,8	5,3 10^{-4}
in all	100	2,9 10^{2}

5 2 3 *Location of fires*

The location of significant fires presented in [6] is shown in Table V One can notice that a great portion of the fires have occurred in turbine generator bearings Also other oil system components and rooms below the main operating floor are often the objects of fires The frequency of fires by location is also shown in this table

Table V. *Distribution of significant fires by location. [6]*

location	percentage [%]	frequency [fires/year]
turbine generator bearings	27,3	7,9 10^{3}
rooms below operating floor	14,5	4,2 10^{3}
oil and steam pipes	10,9	3,2 10^{3}
valves	9,1	2,6 10^{3}
front standard	7,3	2,1 10^{3}
oil tanks	7,3	2,1 10^{3}
turbines of feed water pumps	7,3	2,1 10^{3}
exciter	5,5	1,6 10^{3}
generator	3,6	1,1 10^{3}
switchgears	3,6	1,1 10^{3}
turbine insulation	1,8	5,3 10^{-4}
undetermined	1,8	5,3 10^{-4}
in all	100	2,9 10^{2}

5 2 4 *Operating experience*

At the Loviisa NPP all the four reported events, which have caused a fire alarm, have occurred at the turbine generators two generator carbon brush fires, one small hydrogen fire at the generator's liquid detectors and one smouldering resistor in an exciter room Turbine generators' combined operating time since the year 1996 maintenance outage was 61,4 years So the turbine generator fire frequency is given the value 6,5 10^{2} per year

In order to compare Loviisa's fire frequency to the value $1,1 \cdot 10^{-1}$ taken from [7] we must assume, that the value is the fire frequency of one reactor unit's turbine hall. In that case the corresponding value based on operating experience from Loviisa will be $1,3 \cdot 10^{-1}$ fires per year. Then the frequencies are almost the same.

The first carbon brush fire in Loviisa can be considered significant in terms of [6] because the outage of this turbine generator lasted over one month. So the allegation that a quarter of turbine generator fires is significant seems to be in agreement with the Loviisa NPP operating experience.

The turbine generator systems of all Soviet-designed NPPs are essentially similar to each other. During their operation history there have occurred at least three severe turbine hall fires (Armenia 1, 1982; Beloyarsk 2, 1978; Chernobyl 2, 1991) that required extraordinary and improvised recovery actions in order to prevent core damage. Thus, according to normal PSA methodologies, the core damage frequency caused by severe turbine hall fires in Soviet-designed NPPs could roughly be estimated to be of the order $3...5 \cdot 10^{-3}$ per unit year. However, design improvements and backfittings carried out at several plants as well as improved operation instructions can significantly reduce the risk figure.

6 SUMMARY

As the event frequencies presented earlier in this paper have proved, the most probable consequence of a turbine generator failure is a fire. Because of large fire loads, mainly consisting of lubricating oil, the fire in a turbine hall may grow very serious and may in the worst case make most of the systems situated in the turbine hall inoperable. Other significant fire loads are the hydrogen used for the generator's cooling and the cables and other electrical components, which are located everywhere in the turbine building. Due to high event frequency and serious consequences it can be concluded that the risks of turbine generator failures are considerable.

At a VVER-440 nuclear power plant the risks induced by turbine generator systems are often more significant than at Western 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 at least two units, and there are no walls or fire barriers between the turbine generators and/or safety-related equipment. A turbine generator failure may thus affect the operation of many systems, even the operation of the other unit. Also some other buildings, for instance the control room building and the cooling water pumping plant, can be affected.

Large damages in a VVER-440 turbine hall may affect the primary circuit's residual heat removal from steam generators. The most important safety-related systems located in the turbine hall are the main feed water system, the emergency feed water system, the residual heat removal system and the steam lines from the steam generators, including safety and isolation valves. Further details can be found in the paper "Safety improvements at Loviisa NPP to reduce fire risks originating from turbine generators". As noted in that paper, an independent back-up emergency feed water system and a fire barrier for protection of the control room building and vital parts of the feed water and main steam lines are the most effective and also cost-beneficial way to reduce the risks of turbine hall fires.

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APPENDIX I: TURBINE GENERATOR ACCIDENTS

Table I.I: *Some turbine generator accidents in NPPs*

plant, country, type and date	event description
Armenia-1 Armenia PWR, VVER-440 15 10 1982	A short circuit took place in a 6 kV power cable of a large boron make-up pump Fire started in several places along the cable and caused damage in many power and control cables, which caused several malfunctions Generators connected spuriously to the grid causing several short circuits Turbine generators failed due to electrical and mechanical overload An oil leak set fire to TG2 and its oil tank There was a total station black out Feed water pumping to the steam generators was stopped until the fire in the turbine hall was extinguished after two hours due to the fact that there was only two emergency feed water pumps in the turbine hall
Barseback-1 Sweden BWR 13 4 1979	The retaining ring of a water-cooled generator broke up due to stress corrosion caused by a minor water leak The ring was blown into three pieces which caused extensive damage to the generator, the turbine unit and the turbine hall, like a missile The short circuit ignited leaking lubricating oil causing a fire around the generator and destroying some cables
Beloyarsk-2 Russia LWGR 31 12 1978	A lubricating oil pipe broke in TG2 The oil ignited when it came into contact with hot surfaces on the turbine or steam pipes The fire spread to the cable tunnels and electrical rooms This caused several disturbances and failures and made it difficult to control the plant safety functions
Chernobyl-2 Ukraine LWGR 11 10 1991	TG4 was stopped for minor repairs and adjustment The generator s excitation breaker and air-operated disconnect breakers were opened When the turbine generator had nearly completed its coast down one of the disconnect breakers closed spuriously and reconnected the generator to the grid again The turbine generator accelerated to rated speed within 25 - 30 seconds As a result, the generator rotor got into unbalance and its bearings were destroyed Cooling hydrogen, sealing oil and lubricating oil caught fire All feed water and emergency feed water pumps were lost Lining an alternative make-up water supply and reduction of primary circuit pressure prevented serious consequences
Fermi-2 United States BWR 25 12 1993	An ejected blades from an LP turbine caused damage to turbine building s water lines and to condenser tubes The tube damage resulted in circulating cooling water from Lake Erie being pumped into the reactor cooling system Water from the fire suppression system and the damaged water lines accumulated in the basement of the turbine building and the adjacent radioactive waste processing building There was also a hydrogen fire in the generator's exciter due to breakage of generator sealings Event caused significant damage to the turbine, generator and exciter
Maanshan Taiwan PWR 7 7 1985	Due to torsional vibrations of generator s rotor some LP turbine's blades broke Rotor got into unbalance which caused break down of generator sealings Hydrogen and an oil leak caught fire Generator and exciter were damaged badly
Narora-1 India PHWR 31 3 1993	Cracking of LP turbine blades caused unbalance to turbine generator rotor which led to the failure of bearings A fire was caused by ignition of hydrogen escaping form the generator The oil pipes connected to turbine generator also snapped due to vibrations Fire spread to cable trays in the turbine building and control equipment room, which brought about total station blackout Secondary side cooling was stopped until operators started up the diesel-driven fire pumps in order to feed the steam generators with fire water
Rancho Seco-1 United States PWR 19 3 1984	The loss of the hydrogen side sealing oil pump and other problems in the sealing oil system resulted in the escape of hydrogen from the generator Hydrogen escaped for several minutes before it exploded Following a large explosion the fire burned for five to eight minutes
Salem-2 United States PWR 9 11 1991	During turbine testing at 100 percent power a problem developed in the auto-stop trip system, which led to a turbine and reactor trip The turbine stop valves closed as expected, but after closure they re-opened due to the fact that some solenoid-operated valves of the turbine protection system failed to open Steam flow to the turbine led to an overspeed of about 160 percent of rated speed The LP turbine blades broke and penetrated the turbine shroud A hydrogen explosion and fires and turbine lubricating oil fires resulted The damage in turbine generator systems was extensive, for instance the turbine, the generator and the main condenser were damaged
Vandellos-1 Spain GCGR 19 10 1989	Mechanical failure in the HP turbine blades caused high vibrations in lubricating oil pipes Some pipes broke off and the oil leak caught fire from hot surfaces Also generator's hydrogen leaked out and exploded The fire caused the loss of control air and electrical power of several safety-related components For instance, two turboblowers and main heat exchanges were inoperable The fire also affected a main circulation water pipe expansion joint causing outpouring of sea water which caused the flood of the lower levels of the turbine and reactor buildings

APPENDIX II: TECHNICAL DATA OF TURBINE GENERATORS

This Appendix presents the most important technical parameters of the Loviisa NPP turbine generators. All VVER-440 plants have essentially similar turbine generators.

Table II.I. *Technical data of turbines at Loviisa NPP.*

manufacturer	Harkov turbine factory
type	K-220-44-2
speed of rotation	3000 1/min
shaft power	235 MW
power of HP turbine	105 MW
power of LP turbines	130 MW
pressure of fresh steam	4,3 MPa
temperature of fresh steam	252 °C
flow of fresh steam	350 kg/s
steam pressure after turbine	2,5 kPa
steam temperature after turbine	21 °C
steam flow after turbine	200 kg/s
flow to tapping	150 kg/s

Table II.II. *Technical data of generators at Loviisa NPP.*

manufacturer	Elektrosila, Leningrad
type	TVV-220-2A
speed of rotation	3000 1/min
terminal pair number	1
frequency	50 Hz
terminal voltage	15,75 kV
terminal current	8960 A
apparent power ($\cos\varphi = 0,9$)	245 MVA
active power ($\cos\varphi = 0,9$)	220 MW
reactive power ($\cos\varphi = 0,9$)	108 MVAr
efficiency ($\cos\varphi = 0,9$)	98,7 %

PANEL 2

Panel 2

EXPERIENCE BASED DATA IN FIRE SAFETY ASSESSMENT

Chairperson: **M. Röwekamp** (Germany)

Members: F. Bonino (France)
J. Martilla (Finland)
N. Siu (USA)
G. Wilks (USA)

The panel was opened by the statement of the chairperson that for deterministic as well as for probabilistic fire safety analyses more and more detailed, unbiased data have to be made available to a broad community of fire safety experts including analysts.

The question arose what could be incentives also for nuclear industry to collect and make available further information and how to consider that some of this information has to handled carefully and confidentially. In this context, it was mentioned that the future stronger competition between European utilities could discourage nuclear industry in Europe to share information.

The willingness of the nuclear industry to participate in a broader collection of fire related information and data was stated by one of the panel members as well as the willingness of the participants to make an effort on preparing a common computerised fire related database supported by modern communication systems like email. In this context, there was a common understanding that the data have to be used appropriately avoiding any misuse of the information included and to protect the integrity and sources of data.

As the scientific secretary mentioned, the IAEA is not able to prepare and maintain such a database. Furthermore, the participants expressed their doubts that the agency will only get information via the authorities/governmental institutions of the member states and therefore not always receive all the unfiltered/unbiased information which the experts want to collect.

There was an agreement that knowledge on operating experience of nuclear power plants (NPPs) world-wide is essential and that these data are needed to be collected. Volunteers are requested to build up and maintain such a database.

With respect to minor significant fire related events and precursor events for plant internal fires the panel members recognised difficulties in gaining the information and consistency of this information. The scientific secretary asked the panel members and the participants in the auditory how to get better information and consistency on fire frequencies and their uncertainties.

The analysts clearly expressed their needs for fire ignition models including all types of combustibles and potential ignition sources for characterising fire events

There was a common understanding that there is still a deficiency with respect to the knowledge on cable fires. Further activities including NPP specific tests and fire experiments are needed.

Last not least, panel members as well as several experts from the auditory stated clearly that generic data, in particular those data related to fire protection features, have to be handled carefully, considering differences in the manufacturing as well as in inspection and maintenance, quality control and assurance. With regard to the application of generic fire occurrence frequencies for probabilistic analyses, the available database, mainly from U. S. nuclear power plants, can only be used as a starting point for the plant specific assessment.

FIRE SAFETY REGULATIONS AND LICENSING

(Session 5)

Chairperson

H.P. BERG
Germany

IAEA-SM-345/43

Key Issues Paper

XA9847520

FIRE SAFETY REGULATIONS AND LICENSING

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1. INTRODUCTION

Experience of the past two decades of nuclear power plant operation and results obtained from modern analytical techniques confirm that fires may be a real threat to nuclear safety and should receive adequate attention from the design phase troughout the life of the plant. Fire events, in particular, influence significanty plant safety due to the fact that fires have the potential to simultaneously damage components of redundant safety-related equipment. Hence, the importance of fire protection for the overall safety of a nuclear power plant has to be reflected by the fire safety regulations and to be checked during the licensing process of a plant as well as during the continuous supervision of the operating plant.

However, in early designs of nuclear power plants fire events were not identified according to its actual risk for nuclear safety. Regular industrial standards for fire protection installations were adopted and only somewhat altered to ensure that radiological release was kept within the plant.

The fire at the Browns-Ferry nuclear power plant in 1975 and the development of the general defence-in-depth concept for nuclear power plants have resulted in an improved fire protection strategy which can be achieved by a combination of:

- an adequate design,
- a safety culture, which promotes a proper attitude and ensures continued personnel awareness of potential hazards from fire,
- effective fire prevention and fire protection measures,
- an appropriate level of quality assurance, and
- emergency plans and procedures.

To ensure an adequate fire safety level for all operating plants, the defence-in-depth concept covers the following principal objectives:

- preventing fires from starting,
- detecting fires anywhere in the plant promptly and extinguishing quickly those fires which do start, thus limiting the damage,
- preventing the spread of those fires which have not been extinguished, thus avoiding any common cause failure mode and minimizing their effect on plant safety systems, and
- designing safety systems in such a way that even if a fire spreads within a plant area, the fire will not interfer with the performance of essential plant functions.

In order to protect the health and safety of the public from the potential consequences which a fire may have on reactor safety, each of these principles should meet certain minimum requirements prescribed in regulations or in the license. Strengthening one of these areas may compensate in some measure for weaknesses, known and unknown, in the other areas.

Hence, this new approach is also reflected in more current regulations and in licenses of nuclear power plants of the last generation; however, there was also a need for methods to determine the adequacy of the fire protection programme in plants built to earlier standards and to regulate backfitting measures to enhance the fire safety protection.

Moreover, the knowledge in fire science has increased considerably over the last two decades. Better methods for measuring and analysis have made it possible to look into the basic physics and chemistry of combustion. Modelling of fires has been improved by fluid dynamic science, especially turbulence modelling, and by computer development. This is also a reason of today's trend in some countries to introduce performance-based regulations to replace prescriptive ones.

2. FIRE SAFETY REGULATIONS AND LICENSING

Even in the early years of the design and operation of nuclear power plants, design and regulations have always considered fire safety to some extent. However, nuclear specific requirements and standards related to fire protection were not fully developed.

Nuclear power plants had to meet applicable building codes and standards - the same which were used for other types of industrial facilities and even for office complexes. These non-nuclear regulations contain national and international industrial standards, laws and ordinances for building constructions as well as for fire brigade management and equipment, and ordinances regarding the work place safety. In addition, usually general design criteria and/or safety criteria had described general requirements for fire prevention, detection and response. International guidance is, e. g., provided in [1].

For example in Germany, fire safety is addressed in the safety criterion 2.7 „Fire and Explosion Protection", of the Safety Criteria for Nuclear Power Plants (1977) and the Incident Guidelines of 1982 requiring that protective measures against fires shall be taken by means of plant engineering. The specifications of these precautions were intended to be outlined in four nuclear safety standards describing and prescribing the basic principles, fire safety measures for structural components, fire safety measures for mechanical and electrical components and rescue routes in nuclear power plants. However, only the standard on basic principles has been published. The status of these standards, the benefits and problems with very detailed standards are explained in [2].

Experience shows that very detailed regulations, focused on a specific plant type, are only reasonable if a series of the same plant type is constructed. Otherwise, there appears always the problem how to apply these standards to nuclear power plants built to earlier standards.

These insights advocate that countries establishing or reorganising fire protection regulations should share the experience gained by other regulators.

In [3], general fire protection guidelines for Egyptian nuclear installations are described. Again, the requirements given shall only be applied to new nuclear facilities whereas fire protection modifications of operating plants are considered case by case. This ensures an adequate planning of fire protection measures as an integral part of the design phase and throughout the life of the plant.

Also in the Russian Federation, a revision of the regulatory nuclear framework including fire protection for nuclear power plants has been performed recently and is further developed in 1998 to require a fire safety analysis as part of the operational licensing documents. The fire safety aspects are supervised by the surveillance anthority in close cooperation with a central national fire protection institute which is in the portfolio of the Ministry of the Interior [4].

An important discussion on a fundamental change in fire protection regulations to date has started in the US by proposing a performance-based approach to fire safety and thus making decisions on the basis of fire risk assessments. The regulator's position and the point of view of the utilities are presented in [5] and [6] showing the benefits and problems connected with this new approach. Due to the intended increasing flexibility for the utilities the industry position to maintain the existing regulations and to believe that new rulemaking is not required is an unexpected perspective which is outlined in more detail in [6].

3. KEY ISSUES

Based on past experience, the following key issues for fire protection regulations can be identified:

- laws and ordinances have to define safety goals, general requirements and ways to prove that these goals and requirements are fulfilled,

- standards have to provide the technical basis to achieve an adequate fire safety level, and these standards change with time due to an increase of scientific knowledge and operational experience,

- very detailed and prescriptive standards are helpful in the licensing process because their application simplifies the discussion with the authorities and expert groups and supports possible court decisions regarding the necessary fire safety level of a plant,

- all regulations and standards have their limits; when using them, understanding of their background and context is sometimes of great importance and not a sophisticated interpretation and fulfillment word by word,

- if very detailed fire protection requirements, e. g., on the equipment are prescribed, also the necessary procedures to check the fulfillment should be determined, i. e.

which fire qualification tests are necessary and in which cases qualification by computer codes are sufficient; this is important to make sure that the testing procedure is representative for the application where the system is to be used and that only verified computer codes are applied,

- regulations should not only cover design and construction aspects important in the licensing process but also provide a reasonable procedure for continuous supervision and regular review of the fire safety level of nuclear power plants built to earlier standards on the basis of current standards because a direct application of current standards to older plants is not always suitable,

- modern regulation should take into account the use of probabilistic methods as a supplementary tool to deterministic methods, in particular for evaluating the fire safety level of plants in operation.

4. CONCLUDING REMARKS

Due to the fact that fires are complex phenomena resulting not only from ignition and combustion processes, but also from its impact on safety - related equipment und from the necessary appropriate response of the operators, fires have still to be considered also in modern plants; however, main emphasis remains on nuclear power plants built to earlier standards, in order to ensure the adequate level of fire safety.

In consequence, regulatory anthorities have developed requirements to establish levels of fire safety which must be provided in nuclear power plants. Some of these guidance documents are broad, more performance-based regulations while others are more prescriptive and offer only few choices for the designer, the operators and also the authorities and their experts.

The trend of the US Nuclear Regulatory Commission to change from prescriptive egulations to a risk-informed, performance-based regulation not only in the field of fire protection will be carefully observed and analysed by the European regulators but not followed in the near future. Practical experience will show if this new approach will fulfill the expectations to further improve the fire safety level in a most efficient and cost-effective manner, both for the regulators and the utilities.

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FIRE SAFETY REGULATIONS FOR NUCLEAR POWER PLANTS IN GERMANY AND THE VARIOUS DIMENSIONS OF GERMAN KTA STANDARDIZATION ACTIVITIES. IS THERE A BENEFIT TODAY?

XA9847521

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Abstract

In Germany the mandate for preparing nuclear safety standards is given to the KTA (Nuclear Safety Standards Commission) which has restrictive procedures to definitely ensure consensus principle. The KTA was up to now not in a position to approve comprehensive fire safety relevant standards, although its corresponding program is now 22 years old. KTA 2101.1"Basic Principles of Fire Protection in NPPs" (12/85) is the only one published as valid safety standard. Drafts for 3 additional standards referring fire protection of structural elements, electrical and mechanical components as well as for rescue routes have been agreed upon in working groups, supervised and accepted by the responsible KTA subcommittee, but have not been approved by the full committee of the KTA up to now. Some of these drafts are already more than 5 years old. From the toady's point of view the earliest possibility to have a comprehensive and actual set of fire relevant KTA standards will be in the second half of the year 1999. This would then be 24 years after the first KTA decision to start such a program.

1. German Nuclear Safety Standards Commission (KTA) Legal Background, Organization and Procedures

In Germany the legal basis for the licensing of nuclear power plants and other nuclear installations is the Atomic Energy Act. Amongst others it defines that license can only be obtained „if every precaution which is necessary in the light of existing scientific and technological knowledge is taken in order to prevent damage caused by the construction and the operation of the plant". So in each individual case the nuclear licensing authorities have to decide whether these requirements have been met by the technical equipment or the organizational procedures the applicant intends to install in a nuclear facility. In the early stage of nuclear power technology in Germany licenses were primarily based exclusively on individual evaluation and review, existing technical standards were more or less only valid for conventional that means non-nuclear buildings and did therefore not reflect specific nuclear aspects.

Thus, at the end of the decade of 1960 the need for nuclear safety standards and an elaborating responsible institution was obvious. It is facet of German technical legislation and has long lasting tradition that technical safety standards are developed by commissions which are appointed by Ministers of the Federal Government rather than by governmental agencies or by private institutions. Such commissions were the model for the formation of the German Nuclear Safety Standards Commission, the KTA, which finally was founded in the year 1972 and was at that time given into the responsibility of the German Federal Minister of the Interior. It belongs to the realm of the public law and is indeed neither a governmental agency nor a private organization, like the German Standards Institute (the DIN) or the American Society of Mechanical Engineers (the ASME) or the American Nuclear Society (the ANS).

The commission of the KTA consist of 50 members. Representatives of 5 groups with 10 persons each, who met in former times three time a year, nowadays only once a year. The groups are (1) the manufacturers and vendors of nuclear facilities, (2) the operators/utilities of nuclear facilities, (3) the state licensing authorities and the federal authorities that supervise them, (4) the safety reviewing and advisory organizations and last but not least (5) the other authorities, organizations and institutions dealing with or involved in nuclear technology. The 50 members are nominated for a period of 4 years, the board consists of 5 persons, one of each group. The principle is that the partners cooperate as equal partners and make their decision by consensus. All decisions which are directly related to the establishing of standards must be agreed to by five-sixths of the members who participate in the individual full session of the KTA. This procedure shall ensure that no one of the five groups can be outvoted when a decision has to be taken. Each standard is first published as a draft standard and the public is invited to comment on it within a period of 3 months. Comments received are discussed and taken into account, as appropriate, before then the final version is established and published by the German Federal Minister of the Interior in Federal Gazette.

The preparation of the safety standards is not done by the 50 members of the KTA, but by working groups and supervising subcommittees. They are supported by a scientific technical secretariat. The working group members are experts in the fields of interest and are nominated by the different institutions and companies belonging to the 5 groups of the KTA. The members of the subcommittees are nominated by their institutions , but in addition officially appointed by the KTA. Subcommittees have to supervise and coordinate the activities of the working groups which report to them. The draft standards prepared by the working groups are reviewed by the subcommittees before submission to the KTA. If necessary, the drafts are referred back to the working group and further discussion is initiated. Subcommittees generally have also to review comments received from the public after the publication of the draft standard and generally have to prepare the final version of the standard. So man can imagine that developing a KTA safety standard is rather complicated and slow.

Provisions are also made for to keep published KTA safety standards actual. The subcommittees are responsible for regularly checking the validity of the standards and, if necessary, making recommendations to the commission of the KTA as to alternations deemed necessary to bring a standard up to date.

2. KTA Program and Progress of its Fire Safety Relevant Standards KTA 2101 Part 1 to 3 and KTA 2102

2.1 Initial Program for Fire Protection and Related Activities

To establish uniform fire protection requirements for nuclear power plants and to harmonize the different licensing activities in the various fields and responsibilities involved, the commission of the KTA decided to establish the safety standard set KTA 2101 „Fire Protection in Nuclear Power Plants“ in November 1975. This set was defined to have the following three parts:

Part 1: Basic Principles

Part 2: Structural Elements

Part 3: Mechanical and Electrical Components.

In November 1976 there was an additional KTA-decision to complement the safety standard set KTA 2101 „Fire Protection of Nuclear Power Plants“ by the standard KTA 2102, which should deal with „Rescue Routes in Nuclear Power Plants“ which is a very important fire protection issue to ensure primarily personnel safety. Four different working groups had been established in a relatively short time after, with experts of all the relevant different groups of the KTA. Thus it was ensured that all the different and relevant aspects and implications will be acknowledged early and taken into consideration while drafting the standard. For all these 4 issues so-called preliminary reports with literature studies, collection of licensing experience and first proposals for scope and content have been derived by the working groups and supervised by the responsible subcommittee. These preliminary reports were accepted by the commission of the KTA at its full sessions between February 1978 and October 1979. Thus, at that time it became obvious that there was a common understanding between authorities, utilities, manufacturers and safety reviewing organizations for a realistic possibility for a later acceptance of the different safety standards for fire protection. At that time, at the end of the seventies, the average time period between the first working group meeting and the closing official decision of the KTA for the acceptance of a standard was about 5 years. So the KTA fire protection standard set was ought to be finished by the years 1982/83.

This was not possible due to many reasons. Some are the following ones:

The issue of fire protection is in addition to its nuclear implication an important part of different conventional building laws, which in Germany are per definition independent state laws and are therefor not governed by federal laws, like the Atomic Energy Act. Thus, time consuming discussions have been necessary to harmonize the different strategies between different authorities to reach the goals.

The theoretical concept for the combination of plant-internal respectively -external events with consequential and related fires, which was drafted in KTA 2101.1, was absolutely new and had to be detailed and explained during many discussions. For a fire protection engineer it was at that time very strange to think about design principles for fires in combination with other extremely seldom events.

The new strategy to define necessary fire resistance ratings of structural elements by an analytical method and no more by prescriptive values, coming from conventional building laws and codes, was a little bit like a revolution in German fire design practice. Revolutions normally take time, also in fire protection standardization where in the past most requirements have been based on empirical data.

The scope and content which was originally planned by the working group for „electrical and mechanical components“ was very detailed and therefor opposed those experts who wanted to keep the KTA safety standard more flexible in general than more precise in detail.

But despite all these problems all working groups in charge finally had been able to find common understanding for the requirements on fire protection basic principles (KTA 2101.1), on fire protection of structural elements (KTA 2101.2), of electrical and mechanical components (KTA 2101.3) and on measures to ensure rescue and escape of personnel during fires (KTA 2102). In the course of further discussion the responsible KTA subcommittee also accepted the drafted papers and decided to convey them to the KTA to get the approve as draft standards respectively as standards.

KTA 2101.1 was published as safety standard in **12/85**.

KTA 2101.2 was ought to bee published as draft safety standard in 1993, but was not accepted by the commission of the KTA. It has therefore still the status of a draft safety standard in preparation (issue **06/93**).

KTA 2101.3 was published as draft safety standard in **06/94**. After having discussed comments and proposals from the public it was revised by the working group and accepted by the supervising subcommittee 2 years later in **04/96**. This status still exists up to now.

KTA 2102 was published as draft safety standard in **06/90.** Comments especially from representatives of the German official body, responsible for life and health of working people in general, have led to a revision of the paper, for which the commission of the KTA was asked for official approve as safety standard in **1992**. Up to now no 5/6 consensus was possible within the commission of the KTA in order to publish it as a valid safety standard. Especially since 1992 the operators of the German nuclear power plants argue about the question whether it makes sense to integrate issues of the conventional building code in nuclear safety standards. In this context it is worthwhile to know that in Germany in the sphere of influence of conventional building laws there is an inventory protection for the owners of a building, which does not exist in nuclear affairs where normally he actual state of science and technology form the basis for licensing procedures. This is also the reason why **KTA 2101.2** was not published as official safety standard up to now.

Within the year 1996 - 21 years after the original first decision of the KTA - a group of unshakable optimists (members, especially chairpersons of all the different working groups of KTA 2101) came together to harmonize the different papers, mentioned above. There was an official resolution of the commission of the KTA to do this, in order to get a set of comprehensive requirements at nearly the same level of science and technology, and to come over with the problem of all these different dates of issue. Additionally a small working group of experts who have bee involved in the first preparation of the sole approved KTA 2101.1 „Basic Principles of Fire Protection in NPPs“ (issued 12/85) have met several times in 1996 and 1997 to prepare a revision of the more than 10 years old safety standard which should especially take into consideration the advanced KTA-concept for earthquake design of nuclear power plants and the proposals for the other parts of KTA 2101. Due to general strategies of the KTA for keeping the existing safety standards up to date, the feedback from experience with the existing safety standard is regarded as most valuable and should therefor also be considered during revising an existing standard.

These activities have been completed in 03/97. The result was accepted by the supervising subcommittee in 04/97. Consequently new drafts of harmonized and updated safety standards KTA 2101 part 1 and 2 were again conveyed to the KTA to get the approval as drafts at its session in 06/97. The new draft of KTA 2101.3 was not given to the KTA for the content was already accepted by the KTA recently and was only editorially modified. The subcommittee asked the KTA to approve part 1 and 2, both as draft safety standard, in order to give them to the public and to receive comments from the public.

Both papers had been put on the agenda of the full session in 06/97, but in the very last moment the commission of the KTA decided to delete the topic for **KTA 2101.2** „Fire Protection of Structural Elements in NPPs“, for some members of the KTA had not found enough time to be prepared for being able to vote about its publication as draft standard.

KTA 2101.1 has been in fact discussed by the KTA members, but has not got the necessary 5/6 consensus to be accepted as draft version. It was obvious that the representatives of state licensing authorities refused. Representatives of federal authorities and of the safety reviewing and advisory organizations voted as positive as manufacturers and most of the utilities. But due to the necessary 5/6 consensus they did not succeed. The day after that KTA full session in 06/97 there was an extensive article in a German newspaper entitled ***„Play with fire at the nuclear power plant fire protection"***. In the article it was pronounced that *„ it was indeed a last-minute action of the representative of the state licensing authority of Hessen, where the Biblis npp is located, to avoid that the German utilities and the federal minister responsible for licensing of npps could reduce the fire protection requirements to an unacceptable level"*. For the author and other people it became obvious that discussions about the content of updated fire safety relevant KTA standards seem to be influenced by the different opinions about the role and future of nuclear power in general. Seeing that background it can be argued whether there will be a realistic chance to find a common , that means 5/6 understanding about revised requirements for this issue in the future.

In order to get technically qualified arguments for clarifying the rejection of the above mentioned parties the secretariat of the KTA had now sent all the actual drafts of the 3 parts of KTA 2101 (again) to the different fractions of the KTA shortly after its full session in 06/97. Till the end of October this year (deadline for submission) some comments have been sent to the KTA secretariat, the most extensive one was sent by the German state licensing authority of the state of Hessen, where the 2 elder nuclear power plants at the site of Biblis are located. This comment consists of more than 40 pages.

As man can see, a former huge program for the development of German nuclear safety standards in the fields of fire protection got stuck in the quicksand of higher political decisions. All relevant technical aspects have been agreed upon within the responsible working groups and subcommittees. For the elaborating of the 3 parts of KTA 2101 and KTA 2102- neglecting the preceding time period for the so-called preliminary reports - 164 official working group meeting have taken place, most of them lasted 2 days. At each individual meeting about 15 high qualified experts took part. Due to experience, the preparing time of each participant can be estimated at least three times the course of the working group meetings themselves. So up to now, far more than 10.000 man-hours have been spent by representatives of all the different KTA-groups. Referring that background it is indeed a pity that only one (1!) part of KTA 2101 - the part dealing with basic principles - was agreed upon as valid safety standard. During the eighties the velocity for progress was influenced by the conflicts between existing atomic and conventional requirements, during the nineties it was the conflict between acceptance or rejection of nuclear power in general, also within the different groups of authorities with nuclear responsibilities.

2.2 Some Published or at least Proposed Requirements

2.2.1 KTA 2101.1 „Fire protection in Nuclear Power Plants; Part 1: Basic Principles" (issued 12/85)

All relevant issues are treated with, like e.g.

basic design-philosophy including the special protection of safety relevant features, the overall-postulate of fires in case of the existence of burning material and the consideration of consequential and unrelated fires with plant-internal and -external events as well as

basic design requirements for structural and equipment related fire protection measures and
operational fire protection measures and
tests and inspections.

Referring structural measures it is e.g. stated that basically incombustible construction material shall be used, that fire zones shall be created inside the structures and that the fire resistance rating of walls and ceilings of fire zones shall be at least 90 minutes. Special considerations are included for closures of openings in fire qualified walls and ceilings.

Requirements for equipment related fire protection measures refer e.g. to the special situation inside containment, the design principles for fire alarm and fire suppression systems, for smoke and heat removal as well as for fire protection measures for ventilation systems and off-gas systems.

Referring the issue of tests and inspections for example details for tests prior to the granting of a construction or assembly license are to be seen as well as for the supervision of the construction and for tests prior to the commissioning of the npp and after major repairs and modifications.

2.2.2 KTA 2101.1 „Fire protection in Nuclear Power Plants; Part 1: Basic Principles" (draft version 04/97)

As mentioned in section 2.1 a working group had drafted an updated version of KTA 2101.1 with the aim of primarily harmonizing with the complementary part 2 and 3 but also with the aim to clarify the wording, where in the past misinterpretation and misunderstanding became obvious. That was one of the reasons why the members of that new working group in 1996/97 had been derived out of that former working group which was originally responsible for KTA 2101.1 from the very first beginning. So it was ensured that the original intention was known in the working group.

From the point of view of the working group members (representatives of authorities, safety reviewing companies, utilities and manufacturers) most of the alterations are editorial, but as mentioned in section 2.1, especially one state authority feels that the alterations can lead to an overall degradation of fire safety in German npps in the future. From that the conclusion could be drawn that the clarifications made by the working group and accepted by the responsible supervising subcommittee have met those fields of interpretation-possibilities which are of significance for groups which want to reject nuclear power in Germany.

For further information on the content of the new drafted standards the official secretariat of the KTA, which is part of the „Bundesamt für Strahlenschutz" in D-38201 Salzgitter should be contacted.

2.2.3 KTA 2101.2 „Fire protection in Nuclear Power Plants; Part 2: Structural Elements" (draft version 04/97)

This draft version was agreed upon within the responsible working groups and sub-committee. The most important difference between that draft version (06/93) mentioned in section 2.1 is a totally new analytical concept for the definition of necessary fire ratings of walls and ceilings of fire zones and the closures of openings in these structures. This method, which is based on the actual state of the art in this field of interest in Germany was primarily developed by Prof. Hosser from the University of Braunschweig, Germany and funded by the German federal government. The project management for this development was made by the Gesellschaft für Reaktor- und Anlagensicherheit (GRS) in Cologne which is a nuclear safety reviewing organization.

In this new draft it is required that in cases with a lack of experience the fire proof design of structural elements has to based on either experimental, analytical, plausibility or analogy considerations. When analytical proof is necessary, simplified methods will be allowed. Such a method is part of the draft standard. Necessary fire ratings could now be derived by taking into account real existing fire load, room geometry, ventilation properties, heat sinks, kind of fire extinguishing concept/system and last but not least the safety relevance of the structural element to be designed.

2.2.4. KTA 2101.2 „Fire Protection in Nuclear Power Plants; Part 3: Electrical and Mechanical Elements“ (draft version 04/97)

This draft version agreed upon within the responsible working groups and sub-committee Detailed requirements referring fire detection and fire suppression systems are included as well as specific measures for mechanical components, for electrical equipment and for ventilation systems.

For further information on the content of the actual draft version of this standard the official secretariat of the KTA, which is part of the „Bundesamt für Strahlenschutz“ in D-38201 Salzgitter should be contacted.

2.3 Changing Attitudes in Nuclear Issues and their Influence on Completing the KTA Program

The KTA nuclear safety standards have merely been developed for the design, supervision and license of new nuclear plants. Their application on supervision and control of existing nuclear power plants have not at all been in the center of interest during the first 25 years of KTA working period, beginning in 1972. Within the last 5 -10 years the conviction was increasing in Germany that in the nearer future no new nuclear power plants will be necessary in Germany to guarantee the supply of electricity. Since that time and also in the future activities for design, supervision and license of nuclear facilities will primarily concentrate on the safety assessment of existing elder plants, also in respect to fires. In this context KTA standards can be worthwhile for court proceedings as e.g. anticipated technical opinion of the safety reviewing organizations. For existing elder npp, it is very essential that KTA standards are applied appropriate for they have not been developed for the purpose of reviewing, supervising existing elder plants. That means they have to be applied safety goal orientated, not word by word, without degrading safety level.

This strategy is valid for all safety aspects and especially also for fire safety, for it is fact that the issue of fire safety has not been in the center of interest in the beginning of nuclear power plant design and license about 25 years ago. Within the last 10 years a lot of fire safety reviews and corresponding upgrading activities for elder plants have taken place in Germany. Most of them lasted several years, swallowed a lot of money and were based on the above mentioned safety goal orientated strategy. Only in some cases the applicant for upgrading measures and in some cases also the responsible safety reviewing organization have got troubles with some state licensing authorities, the representatives of which stuck too much on the detailed wording of existing KTA standards.

Those German nuclear power plants which have not been involved in larger fire safety reviewing and upgrading programs within the last 10 years have been designed and licensed during that period where the KTA 2101 safety standard program was developed. Therefore these plants are already designed in compliance with the relevant requirements of KTA 2101.

Within the last decades the KTA standardization activities have mostly been regarded as contribution to ensure a specific safety level within national German boundaries. The last decade dominantly demonstrated that nuclear power will not have an independent national future and that international cooperation will move dominantly forward. Safety standards are an integrated part of this movement. It would be a bad result if the KTA and its members built a blockade on this way.

Beside the national safety relevant standardization activities, new chances for an international future in this field do exist and should be used. One chance will be the program of the IAEA, another chance for the future of standardizing lies e.g. in the bilateral efforts of German and French government to define common safety requirements for future nuclear power plants. Focusing on fire safety a so-called European Technical Code Fire (ETC-F) was prepared by EdF, Framatome, German utilities and Siemens (KWU) within the last 2 years and conveyed to French/German nuclear safety reviewing and advisory organizations in October 1997.

3. European Requirements for a new Type of Pressurized Water Reactor (EPR); Alternative or Amendment to KTA Standards?

As mentioned in section 2.3, within the last 2 years a so-called European Technical Code Fire (ETC-F) was drafted by French and German experts and was handed over for comments and approve to officially nominated safety reviewing organizations in both countries. The drafted ETC-F is based on existing national fire safety regulations or at least drafts of them, like the French RCC-I and the German KTA 2101.1 - 3 and KTA 2102 and is intended to describe that necessary fire safety level which has to be ensured during design of new type of pressurized water reactor, the so-called EPR, which will be built in the future as a cooperation between both countries and will be launched to the international market by about the end of the century.

Contrary to German KTA procedure, but in harmony with the French practice, the ETC-F was exclusively drafted by the industry and then given to nuclear authorities and their consultant. Governments of both countries did accept this strategy. It is too early to conclude that the drafted content will be accepted without any alterations but first discussions show that the basic design relevant fire safety design principles could be agreed upon.

Looking in the future there could be the following understanding:

The requirements of KTA standards describe the necessary safety level of existing German nuclear power plants, designed and licensed exactly at or after the time when the standards have been approved and form therefore the basis for additional reviews and upgrading activities of these plants.

For future new nuclear power plants in Germany the basic design principles are described in the ETC-F. Specific additional national requirements for detail design could be stipulated in the different parts of KTA 2101 and in KTA 2102. So the KTA standards would be amendments and not alternatives to the growing European Technical Code Fire (ETC-F).

4. Conclusion

The German safety standard commission KTA was up to now not in a position to approve comprehensive fire safety relevant standards, although its corresponding program is now 22 years old. KTA 2101.1"Basic Principles of Fire Protection in NPPs" (12/85) is the only one as valid safety standard. Drafts for standards referring fire protection of structural, electrical and mechanical elements as well as for rescue routes have been agreed upon in working groups, supervised by the responsible KTA subcommittee, but not approved by the full committee of the KTA. Some of them are already more than 5 years old.

There seem to be no more interest to realize KTA 2102 as valid standard. As the full committee of the KTA meets only once a year, approved standards for KTA 2101.1 „Basic Requirements" (first revision), KTA 2101.2 „Structural Elements" and KTA 2101.3 „Electrical and Mechanical Components" will not be available until the second half of the year 1999. that means 24 years after the decision of the full session of the KTA to start this program.

The next few months, where again working groups are asked to discuss comments from the different fractions of the KTA, will show whether the members of these groups are furthermore willing to spend their time and money of the companies/organizations they are coming from. Perhaps the activities will be put to an end without having an approved set of fire related KTA safety standards. The existing papers could then at least be regarded as that state of science and technology the responsible working groups have agreed upon. This would at least be worthwhile for education of young engineers and for information of interested companies/organizations from foreign countries.

Countries and governments which in this very moment reflect on establishing an own nuclear safety standardization program should learn about that what is described in this paper. Finding a consensus on technical aspects on behalf of elaborating safety standards is very sensible, especially when commercial markets are no more closed and open for everybody, but it is necessary that the time to find consensus is not too long. To be able to realize this it is necessary generally to restrict on basic principles and not concentrate on detailed requirements. Such requirements should be left into the responsibility of specific licensing and supervising process.

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Invited Paper

TECHNICAL METHODS FOR A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM AT NUCLEAR POWER PLANTS*

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Abstract

This paper presents a technical review and examination of technical methods that are available for developing a risk-informed, performance-based fire protection program at a nuclear plant. The technical methods include "engineering tools" for examining the fire dynamics of fire protection problems, reliability techniques for establishing an optimal fire protection surveillance program, fire computer codes for analyzing important fire protection safety parameters, and risk-informed approaches that can range from drawing qualitative insights from risk information to quantifying the risk impact of alternative fire protection approaches. Based on this technical review and examination, it is concluded that methods for modeling fires, and reliability and fire PRA analyses are currently available to support the initial implementation of simple risk-informed, performance-based approaches in fire protection programs.

1. INTRODUCTION

Historically, requirements for fire protection programs in the general building industry and nuclear power plants have been formulated based on deterministic criteria and prescriptive in nature [1]. In many cases engineering judgment of experts was used in determining fire protection features such as the allowable minimum width of hallways and number of fire detectors and sprinklers for buildings, and the allowable minimum safe separation distance and fire barrier ratings for nuclear power plant safe-shutdown trains. Given the advances in probabilistic risk assessments (PRAs) and fire sciences, fire protection programs can now be revised from being deterministic and prescriptive to more risk-informed and performance-based [1, 2].

In a broad sense, risk-informed, performance-based fire protection programs can be thought of as more efficient in terms of expenditure of resources while at the same time focusing proper attention on risk-significant aspects of the programs. This means is achieved by an increase in risk-informed discrimination offered by PRAs and fundamental understanding of fire dynamics. The two main objectives of risk-informed, performance-based approaches [2, 3] are:

(1) to provide flexibility by emphasizing the safety objective rather than the means for achieving the objective

(2) allocating resources to the most risk-significant areas and minimizing resource allocation to areas in which safety benefit is minimal

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* This paper was prepared by an employee of the United States Nuclear Regulatory Commission. It presents information that does not currently represent an agreed upon staff position. NRC has neither approved nor disapproved its technical content.

In order to achieve the above objectives, it is necessary to establish technical methods that can be used to demonstrate that higher level safety objectives are met. This paper examines and presents technical methods that can be used to implement risk-informed, performance-based fire protection programs.

2. RESULTS OF TECHNICAL REVIEW

In order to determine methods that could be used for risk-informed, performance-based approaches for fire protection, a technical review of the state of the art of fire dynamics and fire probabilistic risk assessments was initially conducted. Some general observations and conclusions from this technical review follow.

The general building industries in several countries (notably New Zealand, Japan, Australia, Canada, and UK) are in a transition from prescriptive to risk-informed, performance--based requirements for fire protection in order to facilitate the approval of innovative designs, reduce costs, and improve safety [4]. The transition taking place is evolutionary in that the prescriptive requirements are still maintained as a frame of reference to determine equivalency, and for approval of standard designs. At the present time, performance-based designs are used for constructing complex new facilities or making extensive modifications to current buildings. The programs initiated toward this goal have required a considerable investment of resources and the development of new engineering talent. Nuclear power plants can benefit from the experience of the building industry in adopting risk-informed, performance-based approaches for nuclear power plants fire protection programs.

The technology of modeling fires (and smoke resulting from fires) is being actively pursued in the general building industry in several countries and nuclear industries in some countries, notably France [5]. Several fire computer codes now available to predict important fire parameters are being validated through international cooperative efforts [6]. The credibility of the results from these codes is dependent on their use within the bounds and in a manner the developers intended. Fire models have been found to be a useful tool for estimating the average thermal environment that causes fire damage.

Several PRA methods and fire computer codes for risk-informed and performance-based evaluations of fire protection alternatives in the general building and nuclear power industries were reviewed. The absolute results of these methods vary significantly because of the uncertainties in the data and models, and because of the manner in which the calculations are conducted, however, the PRA methods are capable of providing useful insights about the relative importance of fire protection features and the risk-significant fire scenarios. Given this current state of the art of fire PRAs, it will be difficult to establish quantitative safety objectives for fire protection programs in such a manner that compliance with these goals can be easily measured. However, information on relative risks can be used with a high degree of confidence. Although certain refinements of fire models and PRAs are desirable, it will be too costly to address all uncertainties to establish quantitative risk goals, and doing so is not essential for initiating applications using results of relative risk. The technology will mature through applications to a stage and time when more sophisticated use of quantitative goals will become feasible.

Fire PRAs conducted in the past have shown that a substantial fraction of the risk from fires in nuclear power plants comes from only three or four areas such as the control room, cable spreading room, and the switchgear room. This risk-information can be used to focus plant resources on these critical fire areas. One means to implement such an approach would be to establish categories, or grades, for the current fire areas in a plant. In such a scheme, a higher level of fire protection would be extended to areas that contribute significantly to plant fire risk.

Changes in core-damage frequency (CDF) calculated in PRAs can also be useful toward determining the safety impact of utilizing alternative risk-informed, performance-based approaches compared to compliance with prescriptive requirements. However, generic conclusions regarding the acceptability of alternative implementation methods are not possible because the results of fire PRAs are dependent on plant-specific compartment and hardware configurations, even if the methods, data, and assumptions are the same. For plant-specific applications, it should be a reliable indicator when the uncertainties in evaluating both the performance-based and prescriptive implementation approaches are similar. Other factors should also be considered in determining the adequacy of alternative approaches, especially if the uncertainties in the analyses for comparing alternative approaches are not similar.

3. SPECIFIC TECHNICAL METHODS AND APPLICATION AREAS FOR LESS PRESCRIPTIVE AND MORE RISK-INFORMED APPROACHES

Based on the above technical review, the following identifies generally categorized specific technical methods that can be used to support a less prescriptive and more risk-informed fire protection program at nuclear power plants. As stated above, given the state of the art of these technical methods, decisions regarding plant fire protection should not be made solely on results from these methods, but these results can be used toward making sound decisions based on performance and risk information. The applications of specific technical methods are presented in order of increasing technical complexity. More detailed examples of some of the applications are presented in the next section. The following list of applications illustrate the applicability of the technical methods and is not intended to be all inclusive.

3.1. Performance-Based Methods

The first general category of methods is those that would support performance-based approaches, but are not necessarily risk-informed, i.e., these methods will support implementation of less-prescriptive safety objectives, but do not directly analyze or utilize risk information.

3.1.1. "Engineering Tools" for Evaluating Fire Dynamics

These "engineering tools" are based on the principles of thermodynamics, fluid mechanics, heat transfer and combustion and are useful for analysis of unwanted fire growth and spread (fire dynamics). These analyses can be mostly conducted by hand without a computer program, or sometimes with simple computer routines of fire correlations. "Engineering tools" are available for calculating an equivalent fire severity, adiabatic flame temperature of the fuel in comparison to the damage temperature of the target, fire spread rate, pre-flashover upper layer gas temperature, vent flows, heat release rate needed for flashover, ventilation limited burning, and post-flashover upper layer gas temperature.

These tools can be used to demonstrate adequacy of deviations from prescriptive requirements for configurations with low fire loading, or to establish the basis for fire barrier ratings, safe separation distance, and need for fire detectors and suppression systems in protecting one train for safe shutdown. Since these tools employ bounding calculations, results will be conservative but can provide useful information to indicate areas where fire protection features have been grossly over-emphasized (or under-emphasized).

3.1.2. Reliability Methods

Feedback of operating experience and reliability modeling techniques can be used to evaluate the performance of alternate fire protection system designs or surveillance schemes. These methods can be used to determine an optimal maintenance and surveillance test interval for fire protection detection, suppression (including fire extinguishers, hoses, and pumps), and lighting systems.

3.1.3. Fire Computer Codes Based on Zone Models

These computer codes are based on plume correlations, ceiling jet phenomena, and hot and cold layer development and can predict the temperature of targets exposed to fires, detector and suppression system actuations, and smoke level and transport during fires. In cases where simple calculations (see *3.1.1*) cannot be used for evaluating fire dynamics to provide useful results (i.e. they are too conservative), these fire computer codes can be used for more detailed calculations to support an assessment of the fire hazard and predicting fire protection system response.

3.2. Risk-Informed, Performance-Based Methods

The second general category of methods is those that would support performance-based and more risk-informed approaches, i.e., these methods will support implementation of less-prescriptive performance criteria, and analyze or utilize risk information.

3.2.1. Use of risk insights in a qualitative manner

The results of PRAs, and other more limited analysis, e.g. using Fire Induced Vulnerability Evaluation (FIVE) method [7] can be used in a qualitative manner to provide risk insights regarding the risk significance or impact of alternate approaches.

An example is the use of fire PRA results, including human recovery modeling, to develop the basis for the plant emergency lighting program in lieu of prescriptive requirements (e.g., eight hours duration for all plant areas containing safe-shutdown equipment). Risk-significant accident sequences, e.g.; for fire induced station blackout, can be examined to determine the need and duration of emergency lighting. In some cases, lighting may be required for more than eight hours.

3.2.2. Risk-Graded Approach

Fire PRA and other methodologies have inherently in them screening processes which can progressively distinguish between and identify high and low risk fire areas. The screening methods employed in fire PRAs, and other methods such as FIVE, can be used toward formulating a risk-graded fire protection program by identifying and focusing on critical fire areas. Categories, or grades, can be established for currently identified fire areas in plants. A higher level of fire protection could then be extended to fire areas that contribute significantly to plant fire risk. An expert panel, consisting of plant fire protection personnel and PRA analysts, should use the results of fire PRAs toward establishing the grades, supplementing the information with engineering judgment, where necessary. This approach would be in contrast to prescriptive requirements that specify that all structures, systems, and components (SSCs) of one shutdown train be protected from fires by the same measures regardless of the extent of vulnerability of those SSCs to a fire or impact on plant risk if they are damaged.

3.2.3. Delta-CDF Calculations

Fire PRA methods can be used to calculate the change in core damage frequency (delta CDF) for alternative approaches to fire protection, including for evaluating the role of operators for recovery actions. These methods are useful for evaluating the extent to which repairs are appropriate to maintain one train of systems to achieve and maintain shutdown conditions, and the use of non-standard systems for shutdown. The methods can also be used to evaluate and compare alternate means of providing fire protection (by combining separation, fire barriers, and detection and suppression) to safe-shutdown systems.

4. EXAMPLES OF APPLICATION OF TECHNICAL METHODS

Trial applications have been conducted or reviewed to evaluate the feasibility of the methods listed above. The following is a discussion of examples of how these methods can be applied to plant fire protection programs.

4.1. Example for Using "Engineering Tools" for Evaluating Fire Dynamics (3.1.1)

In many cases, configurations with low fire loadings (including transient combustibles) can be distinguished from high risk areas through the use of "engineering tools" that represent fire dynamics in a gross manner. The following is an illustration of how simple tools can sometimes be sufficient to predict the degree of threat from fires. A cable spreading room in a nuclear power plant toured by the author is used as an example.

The room is about 6.1 m (20 ft) x 6.1 m x 5.2 m (17 ft) high. The upper half of the room is crowded with cable trays, each of which has an array of cables. There is no observable fuel below the lowest cable tray which is about 3.1 m (10 ft) above the floor. Some cable trays do descend to floor mounted cabinets, but there are only terminal strips in these cabinets, not electrical equipment that could fail and cause a fire. The cables are steel jacketed with no flammable insulation outside the jacket. Although a persistent source of heat could degrade the insulation around individual conductors in the cables, it is unlikely that they can be ignited since air cannot get to the flammable wire insulation.

Since there is nothing combustible in the lower half of the room, a fire can only occur with a "transient" fuel, such as spilled cleaning fluid. Assuming a worst case situation in which the liquid fuel pool is directly below the lowest cable tray, a plume correlation in FPETOOL (a compilation of correlations for fire protection calculations) [8] can be used to estimate the temperature of the plume at the 3.1-m height of the tray for a series of fire sizes. If it is assumed that the wire insulation will start to degrade at 200 °C, and the fuel would burn long enough for the insulation to reach the plume temperature, the corresponding fire size from the correlation is 400 KW. If the fuel is gasoline (most solvents used for cleaning have a significantly lower burning rate than gasoline, e.g., methyl alcohol burns at 1/4 the rate of gasoline), one can use correlations developed for hydrocarbon pool fires [9] to determine that the pool would be about 1.1 m (3.5 ft) in diameter and the liquid surface would burn at about 4.5 mm/minute (7.5 x 10^{-5} m/sec). The volume of the fuel can be determined from the following correlation for the maximum pool diameter.

$$D_m = 2[V^3 g'/y^2]^{1/8}$$

where g' is the effective acceleration due to gravity = 9.8 m/s^2, y = fuel burning rate (m/s)

Solving for V, V = 1.9 x 10^{-4} m^3 = <u>0.2 liters</u>

However, this pool, about 2.5 mm thick, will only burn for about 4 seconds which is insignificant compared to the time that would be required to heat the lowest cable tray to near the plume temperature. These bounding calculations can provide useful information toward plant decisions in terms of the degree of fire protection necessary for different configurations and thermal loads. The tools allow using some information representing the fire dynamics of the problem, and can be used to prevent over-emphasis (or under-emphasis) that can occur when such considerations are omitted and the hazard from all fire areas are equally treated.

4.2. Examples for Using Reliability Methods (3.1.2)

Fire detectors in safety-related areas must be tested periodically, sometimes as frequently as every 3 months in the U.S. Test intervals for all detectors are equally prescribed regardless of performance in codes of the National Fire Protection Association (NFPA), and incorporated into plant technical specifications. Test intervals established based on performance, have been used in other testing programs in nuclear plants [3] and offer opportunities for cost optimization and a focused fire protection surveillance program.

The use of reliability engineering models supported by actual failure data for evaluating appropriate tests intervals for fire detectors has been considered and implemented in a U.S. plant [10]. This plant used fire detector testing records covering a period of five years to establish plant-specific fire detector failure rates. Three types of detectors were considered - ionization, heat, and photoelectric detectors. The surveillance records covered 3 years of semi-annual testing, followed by 2 years of annual testing. Based on the analysis of this performance data, an alternative testing methodology was implemented by the utility by using 10-percent rotating sampling at an annual test interval, with provisions for expanding the sample population if a decline in performance was observed.

More formal reliability methods have been used elsewhere [11] that illustrate the feasibility and benefit of performance-based strategies for fire protection system surveillance. Based on a reliability model, detector failures were classified into random, test-generated, and test-independent faults. Effectiveness of various test strategies for detecting failures was then evaluated, and finally, the parameters of the reliability models (including the uncertainties) were estimated through statistical techniques. These parameters were them included in the reliability model to determine an optimal test strategy. The results indicated that extending the test interval from quarterly to annually, supplemented by daily self-verification and quarterly inspection, would increase the reliability of the detectors and decrease testing costs.

4.3. Example for Using Fire Computer Codes Based on Zone Models (3.1.3)

Fire protection regulations [12] of the U.S. Nuclear Regulatory Commission (NRC) require that one train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage. The regulation provides three options for meeting this requirement including one that allows for separation of cables, equipment, and associated non-safety circuits of redundant safe-shutdown trains by a horizontal distance of more than 6.1 m (20 ft) with no intervening combustible materials or fire hazards. In addition, fire detectors and an automatic suppression system should be installed. Experience from the early 1980's indicated that some utilities in the U.S. found it difficult to implement this prescriptive requirement (it would be too costly), and requested the U.S. NRC that they be exempted from this requirement. In almost all cases, some combination of low combustible loading, a high compartment ceiling, or negligible intervening combustible was used as justification. Utilities indicated that in some cases compliance with the prescriptive requirement would require forced outages and cost up to $ 24 million. The U.S. NRC approved several of the requests for exemptions based on the arguments provided.

Most of the arguments provided by the utilities were qualitative, although at least one utility used correlations in FPETOOL [8] to quantitatively estimate the fire hazard. Since the early 1980's, several fire models and computer codes have been developed that have been used in PRAs and other applications. A study was conducted to evaluate the capability of the following three fire models for developing insights regarding the 20-ft safe-separation requirement: (1) FIVE - a compilation of fire correlations in worksheets for use in screening fire areas [7]; (2) COMPBRN IIIe - a fire computer code developed for fast computations for use in fire PRAs [13]; and (3) CFAST - a fire computer code developed mainly for use in modeling fires in buildings [14].

A representative PWR emergency switchgear room (ESGR) was used for the study.

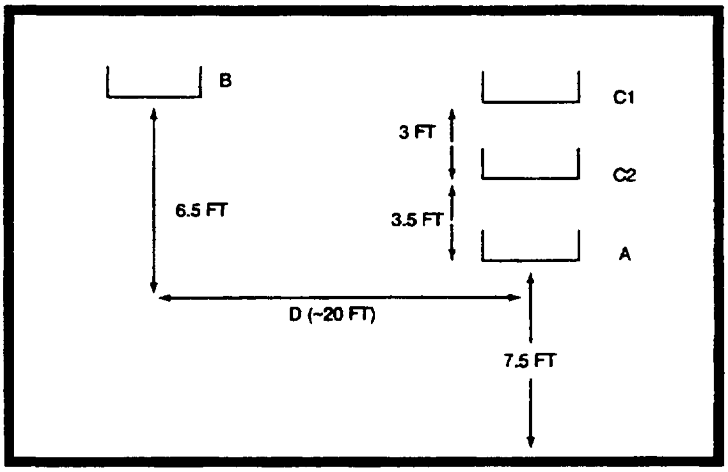

Figure 1. Illustration of Critical Cable Locations in the Representative Emergency Switchgear Room

The room is 15.2 m (50 ft) x 9.1 m (30 ft) x 4.6 m (15 ft) high. The room contains the power and instrumentation cables for the pumps and valves associated with motor-driven auxiliary feedwater trains, all three high-pressure injection trains, and both low-pressure injection trains. A simplified elevation of the ESGR room, illustrating critical cable locations, is shown in Figure 1. The power and instrumentation cables associated with safe-shutdown equipment are arranged in separate divisions and are separated horizontally by a distance, D, in Tray B. The value of D is varied in this evaluation. The analysis was conducted for different elevations of Tray B so that it was either in the ceiling jet sublayer or in the hot gas layer for different cases.

The postulated ignition source is either a self-ignited cable (as a result of a fault) or cable ignition as a result of a transient fire. Cable Tray A is considered to be the source. Although, most rooms will be isolated by the automatic closing of fire dampers and the shutdown of the ventilation system, a small opening 2 m (6.5 ft) high x 0.2 m (0.7 ft) wide was assumed to prevent pressure buildup in the room and facilitate the use of the COMPBRN and CFAST codes.

The ESGR contains smoke detectors and a manually actuated Halon system. Considering the fire initiating frequency and suppression (including fire brigade) probability, it can be estimated that if equipment affecting redundant trains is not damaged within 1 hour, then the resulting core-damage frequency (CDF) for this scenario will be less than 1.2E-5 per reactor-year. This damage frequency and time is used as a measure for determining the adequacy of the safe separation distance.

The FIVE method predicts that an effective fire source intensity of about 6.5 MW is required to damage cables that are separated by 20 ft, and 3.5 MW if separated by 10 ft, for cables that are in the ceiling jet layer (see Table 1). The FIVE screening method does not differentiate between the various separation distances in the hot gas layer and only

Table 1. Summary Results From FIVE Analyses

Effective Fire Intensity KW	Ceiling Jet Temperature K	Target Damage Temperature K	Separation Distance ft
3500	526	643	20
6500	643	643	20
7000	660	643	20
3500	660	643	10
6500	843	643	10
7000	871	643	10

Table 2 Summary of COMPBRN Results

I. Damaged (D) and Ignition (I) Time (minutes)

	Case 1		Case 2		Case 3		Case 4		Case 5	
Tray	D	I	D	I	D	I	D	I	D	I
A (Source)	0	0	0	0	0	0	0	0	0	0
C2	2	2	2	3	2	2	2	2	2	2
C1	4	4	5	5	4	4	4	4	-	-
B (Target)	8	9	9	10	12	No	8	9	No	No

II. Total Heat Release Rate at the Time of Target Damage

	Case 1	Case 2	Case 3	Case 4	Case 5
Q, MW	4.8	4.0	8.2	4.7	1.8*

III. Description of Cases

	Case 1 (Base Case)	Case 2	Case 3	Case 4	Case 5
Pilot fire size (ft x ft)	4 x 2	2 x 2			
Door	Open		Closed		
Trays above pilot fire	C1 and C2				C2 only
Target elevation (m)	4.27			2.29	

* Maximum heat release rate with no damage to target cables

conservatively estimates, based an adiabatic heating of the gas, the total energy release needed to raise the average hot gas layer temperature to the threshold damage temperature. In the present case, the total energy needed is about 286 MJ, which is much less than 3150 MJ corresponding to the energy released from a 3.5 MW fire during a 15-minute period. Therefore, none of the cases pass the screening criteria if the target is the hot layer.

The COMPBRN analyses predict (see Table 2) that the effective fire intensity, capable of damaging redundant cables separated by 6.1 m (20 ft), is about 4 MW for the representative configuration, and that damage occurs in about 12 minutes. The COMPBRN code also predicts that a cluster of two cable trays in one side of the room (Case 5 listed in Table 2) will result in a peak burning rate of about 1.8 MW, which is not sufficient to damage cable trays separated by 20 ft. The heat release rate predicted by COMPBRN for Case 2 is given in Fig. 2.

A modified version of the CFAST code, which accounts for radiation heat transfer to a target, was utilized for this evaluation. The CFAST code requires input of the heat-release rate for the fire source. Values of 1 MW, 2 MW, and 3 MW with a linear growth taking 1, 2, and 3 minutes, respectively for the heat released rate were used for three cases. The hot layer temperature, the radiative and convective heat transfer calculated by CFAST, was used in a transient conduction model for a thin slab to estimate the target surface temperature. Figures 3, 4, and 5 show the hot layer and cable surface temperatures for a 1, 2, and 3-MW fire as a function of time. Considering the critical damage temperature of 643 K and the extrapolation of the result shown in Figs. 3, 4, and 5, a fire of more than 3 MW is required to damage the target cables at a 20-ft separation in less than 1 hour, and a fire less than 2 MW will not damage redundant cables separated by less than 6.1 m (20 ft).

In order to understand the reason for the difference in the predictions of the CFAST and COMPBRN codes, the availability of oxygen to support the burning rates predicted by COMPBRN (see Figure 2) was examined. The CFAST code is capable of calculating the concentration of various species of air and combustible products in the hot layer region, whereas COMPBRN does not account for oxygen depletion and possible starvation of the fire. Using burning rates predicted by COMPBRN, CFAST predicts that, at about 5 minutes, the hot gas layer descends to the level of the lowest burning tray and the concentration of oxygen in the hot layer is below 10 percent (ordinary air is 21 percent). Therefore, the heat release rate will

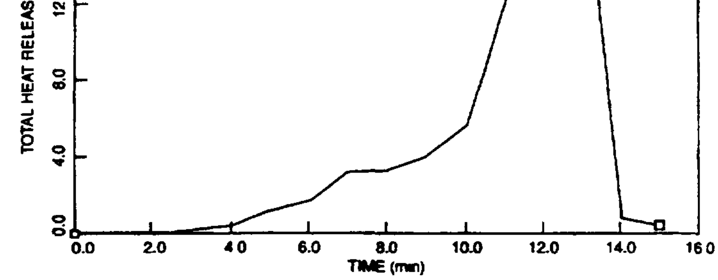

Figure 2. Heat Release Rate Predicted by COMPBRN - Case 2

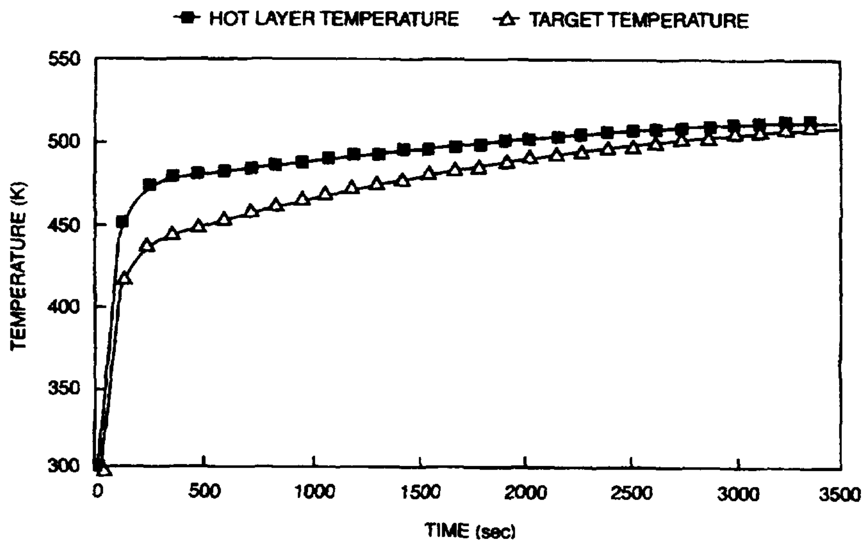

Figure 3. CFAST Prediction of 1-MW Source Target and Hot Layer Temperature

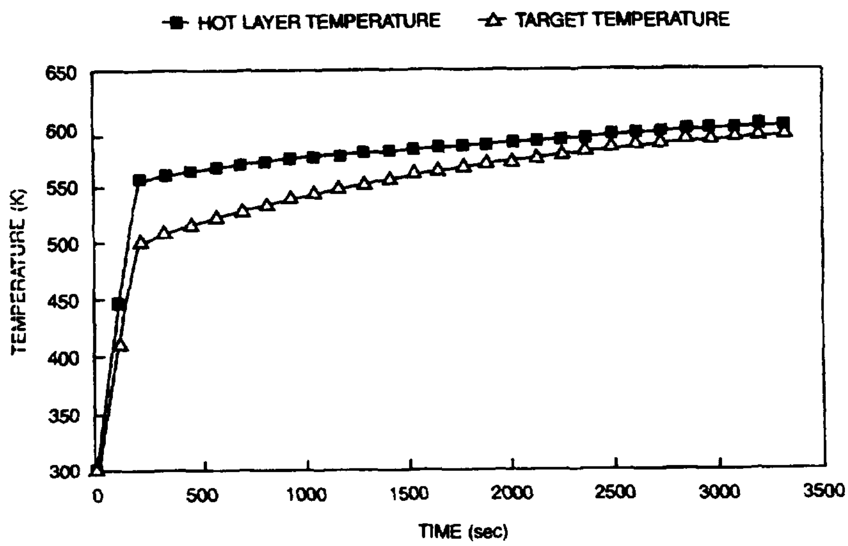

Figure 4. CFAST Prediction of 2-MW Source Target and Hot Layer Temperature

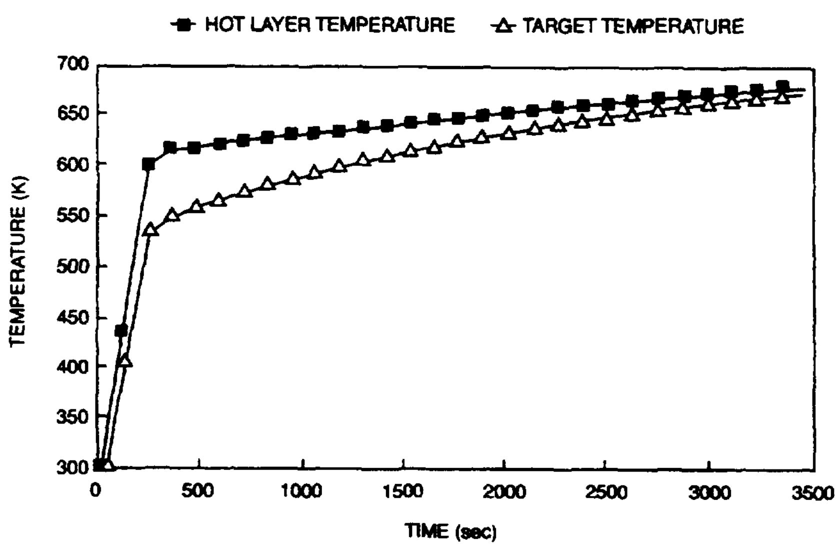

Figure 5. CFAST Prediction of 3-MW Source Target and Hot Layer Temperature

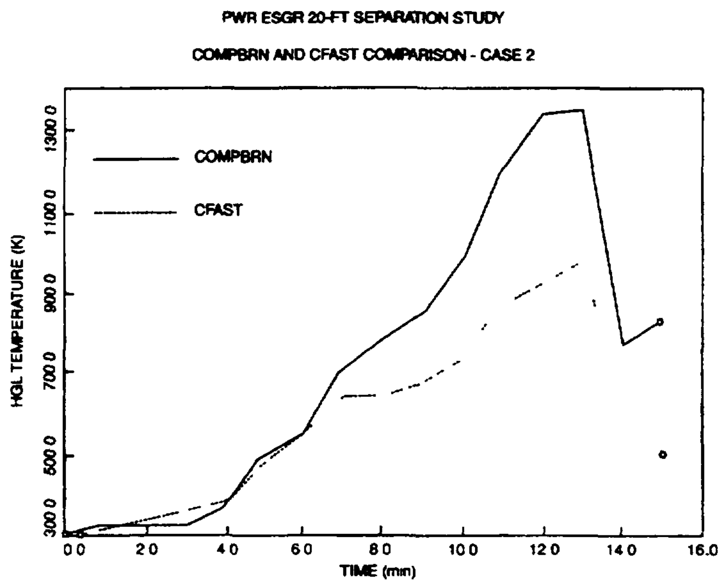

Figure 6. Comparison of CFAST and COMPBRN Prediction of Hot Gas Layer Temperatures

not increase after 5 minutes because of oxygen depletion and the fire would eventually be extinguished when insufficient oxygen is available to support combustion. Accordingly, the peak heat-release rate for this specific case will be below 2 MW and the heat-release rate predicted by COMPBRN after 5 minutes is overly conservative.

Figure 6 shows a comparison of the results from the CFAST and COMPBRN codes for Case 2 (see Table 2 for case conditions). In this case, the heat release rate due to fire predicted by COMPBRN (Figure 2) is provided as input to the CFAST code for the comparison analysis. After the COMPBRN-predicted ignition of Tray C2 at 5 minutes and Tray B (the target tray) at 10 minutes, Figure 6 shows that the hot gas layer temperature predicted by COMPBRN is much higher than that predicted by CFAST. This may be due to the conservative assumptions regarding heat losses from the hot layer in the COMPBRN code, however, the reason for this large difference in hot layer temperature was not examined further.

Based on the above results, it is concluded that if the maximum cluster of source cables results in a heat-release rate less than about 2 MW, then redundant cables will not be damaged, even if they are separated by less than 20 ft (e.g. 15 ft). The dominant factor for all the fire models for predicting damage to cables that are separated by 20 ft is the effective intensity of the fire source, not the total combustible loading in the fire area. Uncertainties in the fire intensity will dominate other uncertainties, such as in calculating the thermal environment, for predictions of cable damage.

The above study illustrates the capability of these fire computer codes to evaluate alternative approaches to the 20-ft separation criteria, although at different levels of resolution The FIVE method is adequate for screening purposes but does not have sufficient resolution to address the problem in this evaluation if it is assumed the target is in the hot layer. The CFAST code provides a better non-conservative estimate for this problem than the COMPBRN code However, both COMPBRN and CFAST estimate that a fire of about 1.8 MW or less will not damage redundant cables with 20-ft separation. This corresponds to a maximum cluster of three cable trays. Although the accuracy of these codes should be improved further, they can already provide approximate results, with a reasonable degree of confidence, that are useful for investigating parameters of interest in fire protection, e.g. the 20-ft separation criteria.

4.4. Example for Using Delta-CDF Calculations (3.2.3)

In order to limit the amount of repairs to equipment for achieving safe shutdown in the event of a fire, current fire regulations of the U.S. NRC require that a plant have the capability to reach cold shutdown conditions within 72 hours [12]. Experience from the early 1980's in implementing this requirement indicates that some U.S. plants found it difficult (it would be too costly) to meet this prescriptive requirement, and therefore requested the U.S. NRC that they be exempted from this requirement based on qualitative arguments that indicated that alternatives that included the use of non-standard systems and repairs, and would require more than 72 hours to reach cold shutdown, would provide an equivalent level of safety. These requests for exemptions based on qualitative arguments were accepted by the U.S. NRC.

Since the early 1980's methods, methods for fire PRAs have become available and can be used to quantify, through delta-CDF calculations, the impact of using alternative methods for achieving the higher level safety objective. The following illustrates this method.

The LaSalle fire PRA analysis [15] for the fire area for the cable shaft room adjacent to the Unit 2, Division 2, essential switchgear room was used for the purpose of this illustration. It was postulated[1] that the fire area contains equipment associated with both trains of the Residual Heat Removal (RHR) System, and that the fire damage is extensive and it will take more than 72 hours to restore one RHR train. This study adopts the LaSalle PRA assumption that a small fire anywhere in the fire subject area will cause the rapid formation of a hot gas layer that causes all critical cabling to fail. Prescriptive compliance with the 72-hour requirement would necessitate that of one RHR train be removed from the fire area, or that it be protected. An alternative approach is postulated to include reestablishing the condenser (Power Conversion System - PCS) for long-term decay heat removal to allow sufficient time for the repair of one train of RHR shutdown cooling. This approach would take more than 72 hours to reach cold shutdown.

The LaSalle fire PRA used conservative assumptions and excluded credit for operator recover actions for modeling the subject fire area since it was a non-dominant contributor to the fire-induced CDF. Therefore a more detailed event tree (shown in Figure 7) was developed for this example which included manual actions to recover PCS and RHR. The prescriptive compliance case assumes one RHR train is removed from the fire area or otherwise protected. Therefore, a failure of the Containment Heat Removal (CHR) function requires additional RHR random failures. The estimated unavailability is CHR = 1.1E-1. The alternative case does not protect the RHR system. All containment heat removal is assumed lost due to the fire, and CHR = 1.0. Operator actions to reestablish the condenser and to recover one train of RHR are critical issues in this analysis. Detailed plant-specific human reliability analysis would be required to accurately represent important operator actions and potential systems interactions. For illustrative purposes, conservative failure estimates were used for these restorations for this study. The four sequences leading to core damage are quantified for both the prescriptive and alternative approaches. The final result is given at the bottom of the Figure; it is ΔCDF = 8.0E-7.

The above example illustrates the PRA method and the feasibility of using ΔCDF as a tool toward evaluating the safety equivalence of an alternative approach to a prescriptive requirement. As is the case for this example, alternate approaches can be expected to require reexamination of non-dominant sequences, and use of a finer level of modeling resolution to credit certain operator recovery actions. The purpose of this example was not to only determine a bottom-line ΔCDF (in any case this analysis is not based on a real plant configuration or

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[1] *It was necessary to assume some changes to the configuration of this fire area in order to allow* data from the LaSalle fire PRA to be used for this illustration. Therefore, this analysis does not model the LaSalle plant.

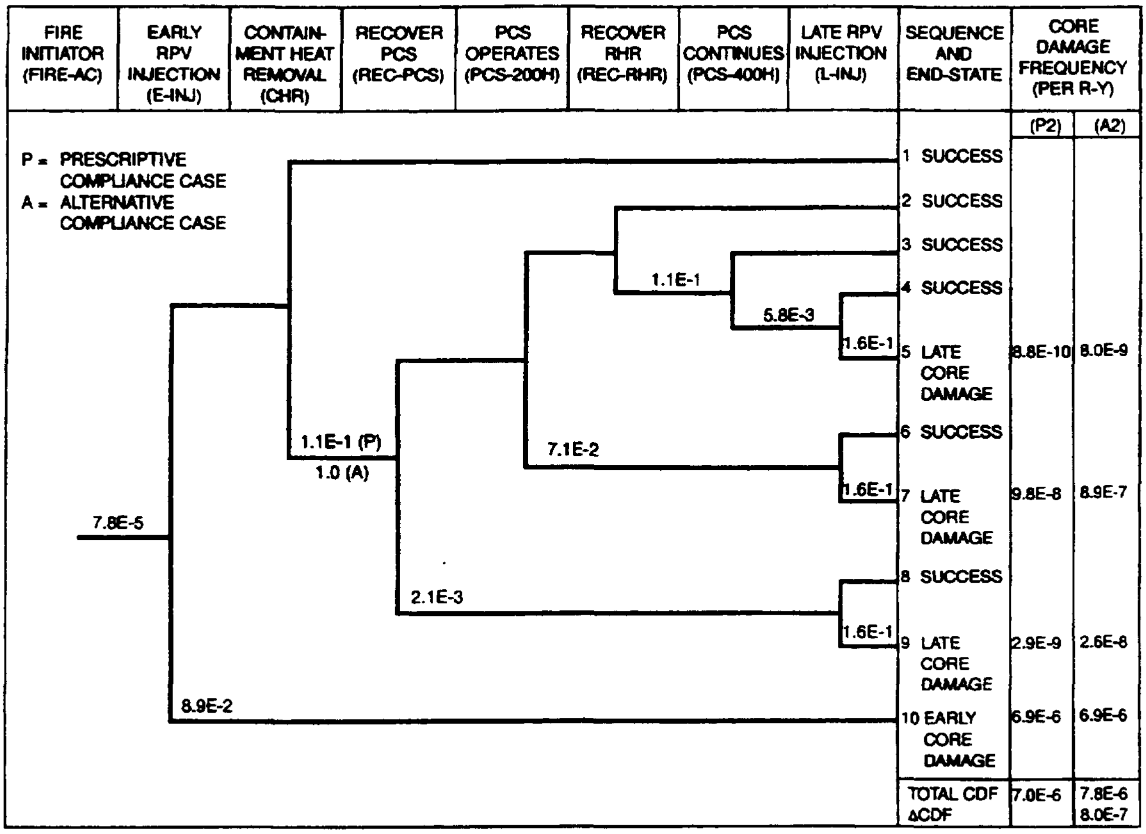

Figure 7. Quantified Event Tree for the 72-Hour Case Study

conditions) but to show that a probabilistic approach provides a consistent framework in which to identify key issues, examine assumptions, sensitivities and uncertainties[2].

5. TECHNICAL EDUCATION AND TRAINING

This paper has presented technical methods that are available for developing a risk-informed, performance-based fire protection program at a nuclear plant. The technical methods include "engineering tools" for examining the fire dynamics of fire protection problems, reliability techniques for establishing an optimal fire protection surveillance program, fire computer codes for estimating fire protection safety parameters, and risk-informed approaches that can range from drawing qualitative insights from risk information to quantifying the effectiveness of alternative fire protection approaches.

Nuclear plant staff that will use the above technical methods will be required to have an adequate level of education and training in these fields. Since fire protection requirements have historically been prescriptive and deterministic, fire protection staff may currently lack the necessary education and skills that are required for accurate and effective use of these methods. Education in the fundamentals of fire dynamics (that mainly includes applications of thermodynamics and heat transfer to fire problems) is necessary to develop the capability to effectively use the "engineering tools" and fire computer codes, and drawing useful and accurate conclusions from such analyses. This field of study has only recently been developed,

[2]The results of the uncertainty analysis for this example is not presented here, but showed that the uncertainty of this analysis is dominated by the uncertainty associated with continued injection after containment failure.

and there are only a few colleges in the U.S. that offer a curriculum that would provide the necessary education. The knowledge and capability to conduct or understand fire PRAs is also normally not possessed by fire protection staff. Training in PRA techniques, including the basics of probability and statistics, will be necessary in order to use PRA and reliability techniques for developing risk-informed, performance-based fire protection programs.

6. CONCLUSION

Methods for modeling fires, and reliability and PRA analyses are currently available to support the initial development of a risk-informed, performance-based fire protection program. Some of these methods require the use of fire computer codes and fire PRAs which will require adequate education and training of the fire protection staff that will use these methods. However, the initial development of a risk-informed, performance-based program does not necessarily require extensive calculations using fire PRAs and models. In many cases, the use of simple performance-based analysis (e.g., "engineering tools" based on fundamental principles of fire dynamics) or application of risk insights in a qualitative manner (e.g., for emergency lighting requirements) is sufficient to examine and implement alternative approaches.

Given the economic status of the nuclear power industry in many countries, including the U.S., the use of risk-informed, performance-based approaches for fire protection programs in nuclear power plants should be implemented in phases. Initially, methods that do not require a significant investment of resources (e.g. in research and training) should be implemented, and the benefits from these applications should be assessed. Based on the assessment of these initial simple applications, the benefits of further investments and development of a risk-informed, performance-based fire program can be evaluated. A phased transition will allow plant resources to be focused for better protection, and the program to be more efficient without the need for a large investment of resources.

ACKNOWLEDGEMENTS

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Invited Paper

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INDUSTRY PARTICIPATION IN THE DEVELOPMENT OF A RISK-INFORMED, PERFORMANCE-BASED REGULATION FOR FIRE PROTECTION AT US NUCLEAR POWER PLANTS

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Abstract

The USNRC staff have recently been directed by the NRC Commissioners to evaluate quickly the development of a risk-informed, performance-based fire protection regulation to replace the current regulations. The US nuclear industry does not believe a new rule is necessary to increase fire safety, and believes that there are significant risks and potentially significant benefits depending on the construction of the rule. However, the industry will actively work with NRC staff if rulemaking proceeds such that the risks are minimized and the benefits maximized. A Nuclear Energy Institute (NEI) Issue Task Force (ITF) has been established to guide the participation of industry in any rulemaking activity. If rulemaking proceeds, a framework should be established for the evolution of a fire protection rule from the current prescriptive basis to a risk-informed, performance-based rule.

1. INTRODUCTION

The resolution of generic fire protection regulatory issues at US nuclear plants is generally coordinated by the Nuclear Energy Institute (NEI). The Fire Protection Working Group (FPWG), comprised of senior industry managers and chaired by a senior utility executive, advises NEI and the industry on the resolution of these issues. [NOTE: The term "industry" refers generically to all US nuclear utilities.] Issue Task Forces of technical experts can be formed by NEI and the Working Group to carry out specific assignments which benefit the industry as a whole. Day to day coordination of generic issues between USNRC and utilities is carried out by NEI staff.

The US nuclear industry does not believe a new rule is necessary to increase fire safety, and believes that there can be significant problems or benefits depending on the construction of the rule. However, if the current NRC rulemaking activities proceed the US nuclear industry has a strong interest in working with the NRC to assure the effectiveness of any new fire protection rule. With its extensive experience in implementing existing rules (10 CFR 50.48 and 10 CFR 50 Appendix R) the industry has much to contribute to the rulemaking process and much to gain from a rule that is more flexible and more easily interpreted and implemented.

The NRC plans to date are outlined in References [1] through [4]. If NRC rulemaking activities continue, coordinated industry activities are expected to proceed in accordance with the plan described below.

2. INDUSTRY PERSPECTIVES ON FIRE PROTECTION RULEMAKING

2.1. Background Information

Specific and detailed requirements for fire protection features and programs are contained in Appendix R to 10 CFR 50. Section II, General Requirements, articulates the following basic elements of the regulation:

- Establish a fire protection program
- Perform a fire hazards analysis
- Establish fire prevention features for those areas containing or presenting a fire hazard to structures, systems and components (SSC) important to safety (specifically those SSC needed to achieve and maintain shutdown conditions)
- Establish alternative or dedicated safe shutdown capability in areas where fire protection features cannot ensure safe shutdown capability.

The regulation is directly applicable to plants licensed prior to January 1, 1979, and was regarded as a physical backfit for those plants (the backfit rule was not in effect at the time). The same requirements (with the exception of III.G.3) were incorporated into NUREG-800, the NRC Standard Review Plan Section 9.5.1 (and hence the FSAR) for plants licensed after this date, based on Branch Technical Position 9.5-1. Plants in the former category must request exemptions, pursuant to 10 CFR 50.12, for any deviations from the prescriptive requirements of Appendix R. (This exemption process has been utilized over 1200 times, making Appendix R the most widely exempted regulation pertaining to nuclear power.) Plants in the latter category may make changes to their fire protection features or programs in accordance with the requirements of 10 CFR 50.59, without the need for the exemption process or advance NRC review. Of course, many aspects of the fire protection program for "Appendix R" plants are subject to change through the § 50.59 process, except for the features that were backfitted by the rule. Additionally, some plants have license conditions addressing fire protection features.

While the regulatory basis for fire protection has remained unchanged since 1981, there has been a significant evolution of NRC expectations with respect to compliance. These have been articulated in a series of NRC Generic Letters, the most significant of which was Generic Letter 86-10 [5], [6]. Actual regulatory oversight has typically been a process of NRC staff interpretations and inspections such that, in the industry view, there has been continuing difficulty in dealing with emerging issues. The process that has evolved is one of making determinations based on utility-specific discussions with the working level NRC staff.

Licensees continue to devote significant resources to modifications to fire protection features and programs in an attempt to stay abreast of technical issues that arise and NRC staff interpretations, examples of which include:

- Protection of safe shutdown pathways through installed hardware (barriers, suppression) versus credit for operator actions
- Acceptable fire tests
- Determination of combustible materials
- Consideration of electrical short circuits due to fire
- Reactor coolant pump lube oil collection systems
- Penetration seals.

A fundamental aspect of Appendix R is that there is often little connection between the actual fire hazard and the level of protection required. (The exemption provision provides the only opportunity to bring actual fire hazards into consideration.) While performance of a fire hazards analysis is mandated, the results of this analysis do not bear on the degree of protection or types of features provided. This was an explicit consideration of the original rule due to difficulty in postulating (with the technology of the time) a "design basis" fire.

As a culmination of several years of efforts, and after receiving public input, NRC recognized the need for revision to Appendix R. In July 1992, NRC issued SECY 92-263, "Staff Plans for Elimination of Requirements Marginal to Safety." [7] NRC recommended initiation of rulemaking to reduce regulatory burden without an adverse impact on safety by making three regulations less prescriptive and more performance oriented. Appendix R was identified as one of these regulations (Appendix J and 10 CFR 50.44 were the others). By letter of August 31, 1992, the NRC Chairman informed the President of the United States of NRC's plans to pursue rulemaking with regard to Appendix R. The NRC Regulatory Review Group (RRG) was formed in 1993 to institutionalize a continuing program of regulatory improvement. This program identified an action plan for rulemaking to replace Appendix R with a performance-oriented, risk-based regulation. This plan identified that industry would petition for rulemaking in this regard. SECY 94-090 [8] established Commission approval of the RRG program.

NEI (then the Nuclear Utility Management and Resources Council) moved forward with plans to prepare a petition for rulemaking. An ad hoc advisory committee was formed to develop an Appendix S to 10 CFR 50 that would provide a performance-based alternative to the prescriptive requirements of Appendix R. The petition was filed in February 1995. Following interactions with the Advisory Committee on Reactor Safeguards (ACRS) and internal review, NRC stated at the Regulatory Information Conference in May 1996 that the "NEI petition does not meet objectives and framework established by the regulatory improvement program." In its response to SECY 96-134, the Commission denied the petition and directed NRC staff to pursue a different approach for fire protection rulemaking, as outlined later in this paper.

2.2. Industry Perspective on Current Regulations

The primary concern of the US nuclear plant fire protection community with respect to the current regulations is the difficulty of consistent interpretation. Since many questions arose in the 1980's about the interpretation of the regulations, the NRC published several guidance documents culminating with Generic Letter 86-10 [5]. Nuclear utilities then used these interpretations to develop fire protection and safe shutdown programs which were later approved (on a plant-specific basis) by the NRC in the late 1980's.

These additional guidance documents have themselves been subject to varying interpretations between industry and the NRC. In some cases, industry representatives thought that they understood NRC intent at the time these guidance documents were developed, only to find that current NRC staff interpret the guidance documents differently. One current example of such differing interpretations is the controversy over how many simultaneous fire-induced spurious equipment actuations must be postulated in protecting a safe shutdown pathway. These varying interpretations can lead either to unintentional violations of the regulation by utilities who thought that they were in compliance, or perhaps to expensive design changes that may or may not meet the intent of the rule. This is not a desirable situation either for industry or the NRC, and efforts are being made to deal with such situations as they arise.

While most industry professionals believe that compliance with the current regulations has been difficult, considerable experience has been developed in applying the current regulations to individual plants. Each plant's compliance with the regulations has been developed over the years through the development of plant-specific fire plans, safe shutdown analyses, NRC approval of formal exemptions to the regulations, and other plant-specific evaluations permitted by Generic Letter 86-10 and 10CFR 50.59. Except for differences in interpretation of the regulations, most plants believe that their fire protection and safe shutdown programs are mature and that changes in the current regulations are not necessary to improve compliance.

2.3. Performance and Risk Basis

NRC management actively encourages wider use of risk-informed and performance-based approaches. Since the initial promulgation of Appendix R there have been significant advances in technology (e.g., fire modeling, risk assessment methods, fire equipment materials and reliability) to support this premise, but it is generally felt that more refinement is necessary before they are used to support a new rule.

A poorly implemented risk-informed or performance-based regulation will only exacerbate the existing problems with regulatory instability and inconsistency. In order for this approach to succeed, it must contain clear objectives, explicit criteria, and a clear tie between fire hazard levels and adequate degrees of protection. Layering of risk and performance requirements on existing requirements will not achieve the benefits desired using this approach.

Establishing useful and quantifiable performance goals relative to fire protection will require careful consideration. The occurrence of fires or of fire protection equipment failures are rare, typically only one or two per year per reactor. Insight that comes from such a paucity of data will be unlikely to provide much flexibility. A better characterization could be the "hazard-based" or "risk-based" measures that can be developed for individual plant fire areas. In the case of fire protection, "performance" may be better characterized by predicting the likelihood and consequences of a rare event, rather than statistically evaluating observed events. Observable measures of performance are still important and can be measured to a greater degree with industry wide experience. Consequently, collecting and interpreting industry fire events data would be an important element of performance based fire regulation.

Measurable fire protection performance means that the regulations, and the degree of protection required, should be tailored to the fire hazard in a particular plant location. Many of the fire probabilistic safety assessments (PSAs) completed to date show that most fire areas are not risk significant, though some results with higher core damage frequencies have caused some NRC concern that compliance with current fire protection regulations does not guarantee low risk levels. Industry firmly believes in the defense-in-depth philosophy, although for certain plant-specific conditions some elements may not be significant to safety. Thus, there could be either a reduction or an increase in fire protection indicated for a particular fire area depending on the PSA results. In general, however, a lessening of requirements could be expected.

A key element is establishing a process to characterize the hazard and associated degree of protection needed. The process for characterizing the potential fire environment is referred to as "fire modeling," and along with PSA is the heart of a performance-based approach to fire protection.

Utilities and NRC have expended significant resources in the development of fire PSAs. Many times, the PSA offers the most detailed model of fire scenarios and the most useful integrated view of fire protection. Efforts are underway to establish criteria for the use of PSA in risk-informed regulation. Building upon the fire PSA tools and results as well as the pilot projects for risk-informed in-service testing and inspections, graded QA and risk based technical specifications, seems a logical step in the development of improved fire protection regulations.

2.4. Industry Perspective on New Fire Protection Rules

Because plant fire protection and safe shutdown programs are at a mature stage, most fire protection professionals are reluctant to undertake a significant shift in the regulations to an untried risk and performance basis. Since there is little experience in the US with regulations based on risk and performance perspectives, industry fire protection professionals are concerned that development of new rules will result in new requirements and new costs that are not justified by improvements in safety. Therefore, a large segment of the US nuclear fire protection community would prefer to maintain the current regulations and methods of compliance rather than spend the time, effort, and expense required to develop and comply with a new rule. Since no new nuclear generation is planned, a new regulation may not be cost beneficial.

In spite of this reluctance to proceed with fire protection rulemaking, there is a recognition of potential benefit from a new rule <u>if the rule is constructed carefully</u>. It would, for instance, be helpful to address emerging generic fire protection issues through consideration of risk significance and cost benefit, thus avoiding costly new requirements with little safety improvement. It would be beneficial to address plant-specific issues through consideration of risk significance or importance using risk or effective fire modeling tools. It could be advantageous to be judged during an NRC inspection on the basis of a fair performance standard rather than prescriptive requirements that may or may not be related to real safety issues. Lastly, it would be of considerably easier to comply with a rule which lends itself to consistent interpretation and compliance.

Given a choice, most industry fire protection professionals would prefer not to proceed with rulemaking. Since there appears to be considerable NRC momentum toward proceeding with fire protection rulemaking, most industry professionals would prefer to see a high degree of industry involvement in any rulemaking activity with the goal of maximizing the benefits and minimizing the problems noted above. NEI has recently requested completion of a survey by all utilities which will provide a clear indication of support for and concerns with rulemaking. The results of the survey will be provided when this paper is presented at the Symposium.

3. INDUSTRY PARTICIPATION IN RULEMAKING

3.1. Industry Goals

If fire protection rulemaking proceeds, it is important to begin any discussion of industry contributions by clearly stating industry goals for any new fire protection rule:

1. Industry should be an integral part of any rulemaking process. The collective experience of the regulated industry should be utilized throughout the rulemaking process. Industry contributions will assist the NRC in assuring clarity in the rule's language and reflecting the perspective of plants that must implement the rule.

2. The rule should be measurable, inspectable, and enforceable. It should be much simpler to determine whether requirements embodied in the new regulation have been met than it is with the current regulations. This approach will help assure improved regulatory stability, including the consistent implementation and enforcement of requirements throughout the industry. The development of simple performance criteria will help meet this goal.

3. The rule should have an option for continued compliance with existing regulations. Since many plants have many years' experience with the current rule and relatively fewer years left in their operating lifetimes, there is no benefit from a regulation which forces backfits in plant fire protection programs. Any additional expense may also decrease the economic viability of some nuclear units. Having an option to continue with existing programs, while offering the flexibility for making changes to those plants that can benefit from them, seems to offer the most promise for regulatory compliance and cost-effective operation.

4. The new rule should offer the capability for evaluating new issues based on safety significance and cost benefit. The new rule should focus on realistic, credible fire initiator and propagation threats and responding to them through good fire protection practices (prevention, detection, and suppression). Some regulatory interpretations of the existing rule require extensive engineering analysis and/or modifications to address very low likelihood fire damage scenarios. The ability to employ a measurable risk criterion should help plant staff focus on the most safety significant fire protection issues. Cost benefit should be considered in conjunction with safety significance in evaluating new issues.

3.2. Industry Contributions to Rulemaking

In the event that rulemaking proceeds, industry has formed an NEI Issue Task Force (ITF) to achieve the following:

- Develop industry goals and performance criteria for supporting a risk-informed, performance-based fire protection rule
- Develop a process for achieving the goals and performance criteria
- Review existing technology and develop recommendations for research or projects needed to support rulemaking

- Manage a pilot program for assessing the value of risk-informed performance-based approaches to real fire protection issues
- Support technical interactions between NRC and industry relative to each of the above activities.

This ITF includes expertise in the areas of fire protection, PSA, plant systems, engineering, fire research (EPRI), regulation, and nuclear insurance. Its first meeting was in September, 1997.

3.3. Industry Position

Based upon the results of a comprehensive survey of U.S. nuclear utilities, the industry has the following positions with respect to the current fire protection rulemaking effort:

- <u>A new rule is not desired, nor is it required to assure or improve safety</u>: Utilities have established safe fire protection and safe shutdown programs under the current regulation. Existing NRC and industry processes have generally been effective in addressing emerging fire protection issues. It is important to preserve the option of compliance with the current rule. Rebaselining existing plant fire protection programs to a new rule would be burdensome, and the safety benefit is not clear.
- <u>Industry will participate extensively if rulemaking proceeds</u>: The fact of 100% utility response to the survey indicates the importance of rulemaking to the industry, which has formed an NEI Issue Task Force to guide industry participation in any rulemaking activity. The industry has many years of experience complying with the current rule, and this experience should be used in making any new rule effective. Industry has projects underway which may be used to support a rulemaking effort, perhaps in cooperation with the NRC.
- <u>The rulemaking schedule must allow adequate time for completion of support elements</u>: The four pilot Fire Protection Functional Inspections and the IPEEEs should be completed to determine if important new safety issues resulting from these evaluations support the development of a new rule. In addition, the activities of the National Fire Protection Association (NFPA) to develop a new performance-based consensus standard for nuclear plant fire protection programs should be considered in rule development, even if the final standard is not complete. Lastly, given the industry's long experience with the exemptions developed to date, and the extent to which these are woven into a plant's licensing basis, any new rule attempting to envelope or supersede them should be carefully considered.
- <u>The use of risk and performance bases has promise</u>: Whether or not rulemaking proceeds, industry sees benefit from the further development of risk and performance tools to support the fire protection regulatory environment. If rulemaking proceeds, it is important to emplace a framework for the further development of these tools; in the absence of rulemaking these tools should be increasingly developed and used within the context of the existing exemption process.

Subsequent steps to support the rulemaking activity will be determined as information about NRC plans unfolds and the responsibilities outlined above are carried out. The NEI Issue Task Force will work with NRC staff as necessary to maintain the NRC's schedule for rulemaking. Industry generally expects to evaluate risk-informed, performance-based approaches to fire protection through practical examples as rulemaking proceeds. This approach should help focus any needed research or development activities on those areas where current technology is not adequate, and help assure the final usefulness of the rule while it is still being developed.

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РЕГУЛИРУЮЩАЯ ДЕЯТЕЛЬНОСТЬ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ АЭС В РОССИЙСКОЙ ФЕДЕРАЦИИ: ЗАКОНОДАТЕЛЬНАЯ БАЗА И ОПЫТ ЛИЦЕНЗИРОВАНИЯ

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Abstract-Аннотация

REGULATORY ACTIVITIES IN THE FIELD OF FIRE SAFETY AT NUCLEAR POWER PLANTS IN THE RUSSIAN FEDERATION: LEGISLATIVE INFRASTRUCTURE AND LICENSING EXPERIENCE.

This paper discusses regulation of fire safety at nuclear power plants in Russia, including issues relating to the legislative infrastructure and licensing activities. The interaction of regulatory bodies in Russia (Gosatomnadzor and the State Fire Fighting Service of the Russian Ministry of Internal Affairs) on the issue of fire safety at nuclear power plants is discussed in detail. The legislative infrastructure for regulation of fire safety at nuclear power plants is described and ways of improving it are discussed, as well as the development of a new fire protection approach, fire protection programmes with acceptability criteria, and methodologies for analysing the effect of fires on nuclear power plant safety. The process for detecting and eliminating faults in the regulation and implementation of fire safety measures at nuclear power plants is also described. In addition, we take a new look at the activities of Gosatomnadzor - licensing, including fire safety at nuclear power plants.

РЕГУЛИРУЮЩАЯ ДЕЯТЕЛЬНОСТЬ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ АЭС В РОССИЙСКОЙ ФЕДЕРАЦИИ: ЗАКОНОДАТЕЛЬНАЯ БАЗА И ОПЫТ ЛИЦЕНЗИРОВАНИЯ

Данный доклад раскрывает вопросы регулирования пожарной безопасности (ПБ) на АЭС России, включая вопросы законодательной базы и лицензионную деятельность. Подробно изложены вопросы взаимодействия регулирующих органов России - Госатомнадзора России (ГАН РФ) и Государственной противопожарной службы Министерства внутренних дел России (ГПС МВД РФ) по вопросам ПБ на АЭС. Представлена законодательная база в области регулирования ПБ на АЭС, излагаются вопросы ее совершенствования, разработки новой концепции по противопожарной защите, программ противопожарной защиты с критериями приемлемости, а также методик по выполнению анализа влияния пожаров на безопасность АЭС. Изложен также процесс выявления недостатков и их устранения в системе регулирования и обеспечения ПБ АЭС. Представлен также новый вид деятельности ГАН РФ - лицензирование, включая вопросы ПБ АЭС.

1. ОБЩАЯ ЧАСТЬ

1.1. Организация и регулирование противопожарной безопасности АЭС государственными органами России.

Современное состояние и развитие атомной энергетики в России неразрывно связаны с чернобыльской аварией. После первой тяжелой аварии на атомной электростанции "Три МАЙЛ Айленд-2" в США (1979 г.) во многих странах мира, развивающих атомную энергетику, началось переосмысление подходов к обеспечению безопасности атомных электростанций. К сожалению в Советском Союзе эта работа проводилась недостаточно интенсивно и только после чернобыльской аварии изменился концептуальный подход к обеспечению ПБ. Уже в 1988 г. была закончена разработка нового регулирующего документа - "Общие положения обеспечения безопасности АЭС" (ОПБ-88), определившего на тот период концепцию безопасности атомных станций.

Одновременно с внедрением концепции безопасности АЭС в России существенные изменения произошли и в подходах к надзору и регулированию ПБ на АЭС. Усовершенствовалось государственное нормирование в области обеспечения ПБ путем установления норм, правил, руководящих, распорядительных и иных документов. Совершенствовалась разрешительная и инспекционная деятельность.

В настоящее время на ГАН РФ возложены функции не только осуществления надзора , но и регулирования
безопасности (включая ПБ) на АЭС, а также во всей ядерной отрасли, включая транспорт ядерных материалов и предприятия топливного цикла.

Законодательство Российской Федерации о пожарной безопасности АЭС основывается на Конституции РФ и включает в себя Федеральные законы "Об использовании атомной энергии"(1995г.) и "О пожарной безопасности" (1994г.).

Важным условием обеспечения ПБ на АЭС является выполнение требований норм и правил по данному направлению, хотя пожарная защита действующих АЭС России, которые были введены в эксплуатацию с 1971 по 1994 годы., проектировалась и строилась по общепромышленным нормам и правилам и не учитывалась специфика влияния пожара на состояние ядерной и радиационной безопасности АЭС. Позже был разработан и введен ряд документов, основанных на требованиях ОПБ-88, регламентирующих проведение анализа безопасности при возникновении пожаров на АЭС.

Законом "Об использовании атомной энергии" определена компетенция Федеральных органов исполнительной власти, организаций и предприятий по вопросам организации и регулирования ПБ на АЭС, а именно:

- обеспечение и контроль за ПБ на АЭС возложен на эксплуатирующие организации (Минатом России);

- государственное регулирование и надзор за ядерной и радиационной безопасностью на АЭС возложены на ГАН РФ;

- государственное регулирование и пожарный надзор за соблюдением на АЭС требований ПБ возложены на МВД РФ.

Основой регулирования государственными органами вопросов ПБ на АЭС в Российской Федерации является разрешительная деятельность (лицензирование, контроль, санкции).

В 1997 году Правительством РФ было введено в действие "Положение о лицензировании деятельности в области использования атомной энергии", которое устанавливает для ГАН РФ порядок и условия лицензирования деятельности эксплуатирующих организаций АЭС. Этот документ является для ГАН РФ основополагающим, в соответствии с которым ГАН РФ в настоящее время пересматривает (наработанный ранее) пакет руководящих документов по вопросам лицензирования в области использования атомной энергии.Со второй половины 1997 года ГАН РФ выдает эксплуатирующим организациям лизензии на вид деятельности (взамен временных разрешений). Аналогичное "Положение..." введено в 1996 году и в системе МВД РФ.

В целях взаимодействия и разграничения компетенций в области надзора за обеспечением ПБ ядерно- и радиационно-опасных объектов (в первую очередь АЭС) между ГАН РФ и ГПС МВД РФ было заключено (1994г.) соответствующее соглашение. Основными принципами взаимодействия являются:

- выработка единых позиций и взаимное согласование решений в области обеспечения ПБ;

- проведение совместных мероприятий по совершенствованию требований противопожарных норм и правил.

В указанном соглашении также определены основные направления взаимодействия, в частности:

- разработка законодательных и иных нормативных актов по проблемам ПБ ядерно- и радиационно-опасных объектов;

- организация и проведение научных исследований в области совершенствования требований ПБ;

- изучение, анализ пожаров и разработка на этой основе предложений по внесению изменений и дополнений в нормативную документацию;

- проведение совместных проверок деятельности проектных организаций, ядерно- и радиационно-опасных объектов на всех этапах жизненного цикла и пр.

В этом же документе разграничены полномочия сторон следующим образром:

а) ГАН РФ:

- утверждает нормы и правила по безопасности при использовании атомной энергии (по согласованию с ГПС МВД РФ);

- проводит экспертизу проектных и иных материалов и дает оценку достаточности принятых решений для обеспечения ядерной и радиационной безопасности в случае пожара;

- выдает лицензию на эксплуатацию объектов, а также разрешение на пуск блоков после проведения плановых ремонтов (с учетом мнения органов ГПС МВД РФ) и пр.

б) ГПС МВД РФ:

- утверждает нормы и правила по ПБ для ядерно- и радиационно-опасных объектов (по согласованию с ГАН РФ), содержащие требования ПБ;

- осуществляет государственный надзор за соблюдением на вышеуказанных объектах требований норм и правил в области ПБ;

- участвует в работе комиссий по приемке в эксплуатацию объектов и систем противопожарной защиты, включая системы обеспечивающие безопасность;

- ведет учет пожаров и информирует ГАН РФ о противопожарном состоянии объектов и результатов инспектирования и пр.

Данное соглашение между ГАН РФ и МВД РФ в настоящее время выполняется обеими сторонами.

1.2.Нормативно-техническое обеспечение противопожарной защиты АЭС.

Важным условием обеспечения ПБ на АЭС в значительной степени является выполнение эксплуатирующими организациями Минатома России требований норм и правил в данной области.

Противопожарная защита действующих АЭС России, как указывалось выше, проектировалась и строилась по общепромышленным нормам и правилам, когда еще не было отдельных нормативов по ПБ.

В 1985 году было введено "Типовое содержание технического обоснования атомных станций" (ТС ТОБ АС-85), которое требует проведения анализа безопасности при пожарах в помещениях АЭС с учетом выхода из строя всех находящихся в них элементов или каналов безопасности.

В 1988 году были введены "Противопожарные нормы проектирования АЭС" (ВСН 01-87).

После ввода ОПБ-88 документы, содержащие требования к проектированию и строительству противопожарной защиты АЭС, не пересматривались и вновь не выпускались, за исключением "Правил пожарной безопасности при эксплуатации АЭС" (ППБ АС-95), которые были введены в 1995 году Минатомом России.

Таким образом, назрела необходимость создания нового пакета нормативных документов по регулированию и надзору за противопожарной защитой действующих и вновь строящихся АЭС, в том числе, с использованием как отечественного, так и зарубежного опыта детерминистского и вероятностного подходов к анализу ПБ на АЭС.

Намечены пути реализации данной проблемы. В частности, ГАН РФ решением коллегии в 1996 году обязал МВД РФ (при участии ГАН РФ) разработать, с учетом накопленного зарубежного опыта, концепцию противопожарной защиты АЭС, в основу которой должны быть заложены принципы безопасного останова, гарантированного расхолаживания реакторной установки (РУ) и обеспечения локализации и контроля радиоактивных выбросов в окружающую среду. Основная цель концепции - разработка отраслевого нормативного документа, учитывающего накопленный российский и международный опыт и, в концептуальном виде, излагающего современные методы и подходы к вопросам ПБ АЭС. Концепция будет содержать более высокие и современные требования к вопросам ПБ АЭС и, в первую очередь, к анализу возможности безопасного останова РУ в случае пожара. Такой анализ будет заключаться в том, чтобы документально обосновать достаточность противопожарной защиты для безопасности АЭС.

Концепция противопожарной защиты на АЭС России будет учитывать требования выпущенных ранее нормативных документов, а также будет содержать следующие основные положения:

- требования к вновь разрабатываемым нормативным документам по обеспечению ПБ;
- требования по объему глубокоэшелонированной пожарной защиты АЭС;
- требованию по проведению анализа возможности безопасного останова энергоблока в случае пожара во всех режимах эксплуатации;
- требования по разработке всеобъемлющей программы противопожарной защиты для каждой АЭС;
- требования к системам резервного управления и расхолаживания;
- требования к системам обнаружения и тушения пожаров;
- требования к обучению и тренировкам персонала и пр.

В качестве исходных материалов используются документы МАГАТЭ, а также документ Министерства энергетики США "Методология по оценке пожарной безопасности на АЭС с ВВЭР и РБМК (RCREM)".

В рамках разрабатываемой концепции эксплуатирующими организациями разрабатываются программы противопожарной защиты по каждой АЭС. Задача программ противопожарной защиты заключается в том, чтобы гарантировать возможность остановки реактора и поддержания его в безопасном состоянии останова, а также возможность сокращения до минимума утечки радиоактивных выбросов в окружающую среду в случае возникновения пожара.

Эти программы включают в себя комплекс мероприятий, охватывающий компоненты, процедуры и персонал, необходимые для осуществления всех мероприятий противопожарной защиты.

Программы противопожарной защиты будут удовлетворять следующим основным критериям приемлемости:

- сооружения, системы и компоненты, существенные для безопасности будут проектироваться, конструироваться и размещаться с таким расчетом, чтобы свести к минимуму вероятность пожаров и взрывов, а также возмещенный ущерб от них, не поступаясь при этом другими требованиями в области безопасности;

- эксплуатация систем обнаружения пожара и противопожарной защиты, а также административный контроль будут направлены на защиту конструкций, систем и компонентов реактора, связанных с обеспечением безопасности;

- конструкции, системы и компоненты, связанные с обеспечением ПБ будут гарантировать их способность выполнять предусмотренные для них функции;

- персонал АЭС будет соответствующим образом подготовлен в части обеспечения ПБ.

Эксплуатирующим организациям предложено также разработать методику выполнения анализа влияния пожаров и их последствий на безопасный останов РУ, а Минатому России пересмотреть до 2000 года действующую нормативную базу в соответствии с указанной концепцией. Главная задача методик заключается в проведении анализа противопожарной защиты для безопасности АЭС.

Кроме того распорядительным документом по Минатому РФ предписано эксплуатирующим организациям выполнить анализ состояния ПБ для каждого блока АЭС (на основе определения риска возможной тяжелой аварии) и по результатам проведенного анализа разработать соответствующие мероприятия по повышению состояния ПБ на АЭС. Такой анализ уже выполнен по блоку 3 Балаковской АЭС (ВВЭР-1000) и (при финансовой поддержке Министерства энергетики США) по блоку 3 Смоленской АЭС (РБМК-1000).

В соответствии с “Положением о лицензировании в области использования атомной энергии” эксплуатирующими организациями направляются в ГАН РФ, в составе представляемых документов для получения лицензий на эксплуатацию блоков АЭС, отчеты по противопожарной защите с анализом влияния пожаров на выполнение функций безопасности и с информацией по реализации мероприятий по устранению недостатков, выявленных при анализе.

В приложении 1 приведена структура законодательной базы и нормативно-правового обеспечения в области регулирования пожарной безопасности АЭС России, а в приложении 2 приведен перечень основной нормативной документации ГАН РФ, используемой при надзоре и регулировании вопросов ПБ на АЭС.

1.3. Процесс выявления регулирующими органами недостатков в системе обеспечения противопожарной безопасности.

Выявление недостатков в системе обеспечения ПБ на АЭС ведется регулирующими органами на всех жизненных этапах АЭС. В частности, ГАН РФ, выполняя свои регулирующие функции, проводит (перед выдачей лицензии) экспертизу материалов обосновывающих ядерную, радиационную и пожарную безопасность. Особое внимание при этом уделяется проектным материалам, в которых закладываются основы ПБ. Экспертное заключение по рассматриваемым материалам является основой для составления условий действия лицензии на вид деятельности, куда входят также требования ГАН РФ об устранении выявленных недостатков в области ПБ.

Основным документом, которым в настоящее время руководствуется ГАН РФ при выявлении недостатков в системе обеспечения ПБ на АЭС и оценке влияния *пожара на выполнение функций безопасности является “Техническое обоснование* безопасности АЭС” (ТОБ АС), имеющий разделы по анализу воздействия пожара на технологическое оборудование и системы безопасности.

Непосредственно на АЭС региональными органами ГАН РФ выдаются поэтапные разрешения на проведение ремонтных работ, работ по модернизации оборудования и систем и пр., при которых ГАН РФ выставляются условия к эксплуатирующим организациям об устранении в определенный срок выявленных (имеющихся) отступлений от норм и правил по обеспечению ПБ на АЭС.

Другим видом выявления недостатков в области обеспечения ПБ АЭС являются инспекции (комплексные, целевые и оперативные) государственных регулирующих органов (ГАН РФ и ГПС МВД РФ). В программы таких инспекций включаются вопросы проверки выполнения эксплуатирующими организациями требований по наличию документов об организации противопожарной защиты, подготовки персонала и выполнению противопожарных мероприятий.

По результатам выявленных нарушений требований нормативной документации регулирующими органами выдаются руководителям АЭС предписывающие документы для устранения недостатков в области обеспечения ПБ.

1.4. Опыт лицензирования АЭС.

Создание и реализация разрешительной системы при использовании атомной энергии является одним из главных направлений деятельности ГАН РФ. Становление разрешительной деятельности ГАН РФ осуществлялось в условиях отсутствия в России соответствующего законодательства об использовании атомной энергии.

При создании разрешительной системы ГАН РФ опирался на:

- полномочия, предоставленные ГАН РФ Президентом Российской Федерации по выдаче разрешений на виды деятельности;

- перечень видов деятельности, на которые субъекты предринимательской деятельности и предприятия, независимо от форм собственности, должны получать разрешения (лицензии).

Перечень видов деятельности включает:

- строительство, ввод в эксплуатацию, эксплуатацию, реконструкцию (включая модернизацию и ремонт в течение срока службы), снятие с эксплуатации ядерно- и радиационно-опасных объектов;

- производство, транспортирование, переработку, хранение, захоронение и использование ядерных материалов, радиоактивных веществ и изделий на их основе;

- хранение, переработку, транспортирование, утилизацию и захоронение радиоактивных отходов.

Первым шагом по созданию разрешительной системы со стороны ГАН РФ была разработка и введение в действие "Положения о порядке выдачи временных разрешений на эксплуатацию блоков атомных станций" (1993г.).

В 1994 году было разработано и введено в действие "Положение о порядке выдачи временных разрешений ГАН РФ на строительство блоков атомных станций гражданского назначения" и ряд других руководящих и методических документов по организации и проведению экспертизы. Данный документ устанавливает порядок подачи заявки со стороны эксплуатирующей организации в ГАН РФ, требования к объему документации по обоснованию безопасности, процедуре рассмотрения материалов и выдачи разрешения на эксплуатацию.

Процедурными документами ГАН РФ предусмотрено, что при выдаче временных разрешений на строительство и эксплуатацию блоков АЭС проводится их оценка безопасности (включая вопросы ПБ), которая выполняется путем анализа соответствующих обосновывающих документов и проведением, в случае необходимости, инспекций как самого блока АЭС, так и предприятия, претендующего на роль эксплуатирующей организации.

Порядок осуществления регулирующей деятельности ГАН РФ при проведении владельцем модернизации АЭС (включая вопросы обеспечения ПБ) аналогичен вышеописанным процедурам с той лишь разницей, что решение владельца о проведении модернизации сначала согласовывается с ГАН РФ и только после разработки мероприятий по реализации соответствующего технического решения, разработки проектно-конструкторской и технологической документации, изготовления соответствующего оборудования, выполнения необходимых строительно-монтажных и пусконаладочных работ на энергоблоке АЭС, корректировке эксплуатационной документации - ГАН РФ выдает разрешение владельцу на ввод энергоблока АЭС в эксплуатацию.

При проведении анализа документов, обосновывающих ядерную, радиационную и пожарную безопасность ГАН РФ опирается на поддержку созданного при ГАН РФ научно-технического центра по ядерной и радиационной безопасности (НТЦ ЯРБ), на кототрый руководящими документами ГАН РФ возложена обязанность по организации и проведению экспертиз безопасности ядерно- и радиационно-опасных объектов, в том числе с привлечением внешних специалистов и организаций.

Первая заявка на получение разрешения на эксплуатацию блока 3 Смоленской АЭС по указанной процедуре поступила в ГАН РФ от эксплуатирующей организации концерна "Росэнергоатом" в конце 1993 г. После рассмотрения материалов в январе 1995 г. ГАН РФ выдал разрешение на эксплуатацию блока. В настоящее время выданы разрешения на эксплуатацию 10 блокам АЭС, а также ведется рассмотрение еще ряда заявок от эксплуатирующих организаций.

Случаев приостановки со стороны ГАН РФ действия разрешений на эксплуатацию блоков АЭС по причине, связанной с обеспечением ПБ, не было. Региональными структурами ГАН РФ и ГПС МВД РФ многократно приостанавливались работы и эксплуатация различных помещений и оборудования, находящегося в пожароугрожаемом состоянии. Например, только в 1996 году государственными регулирующими органами была приостановлена работа и запрещена эксплуатация помещений или оборудования (изымались разрешения) в 995 различных случаях.

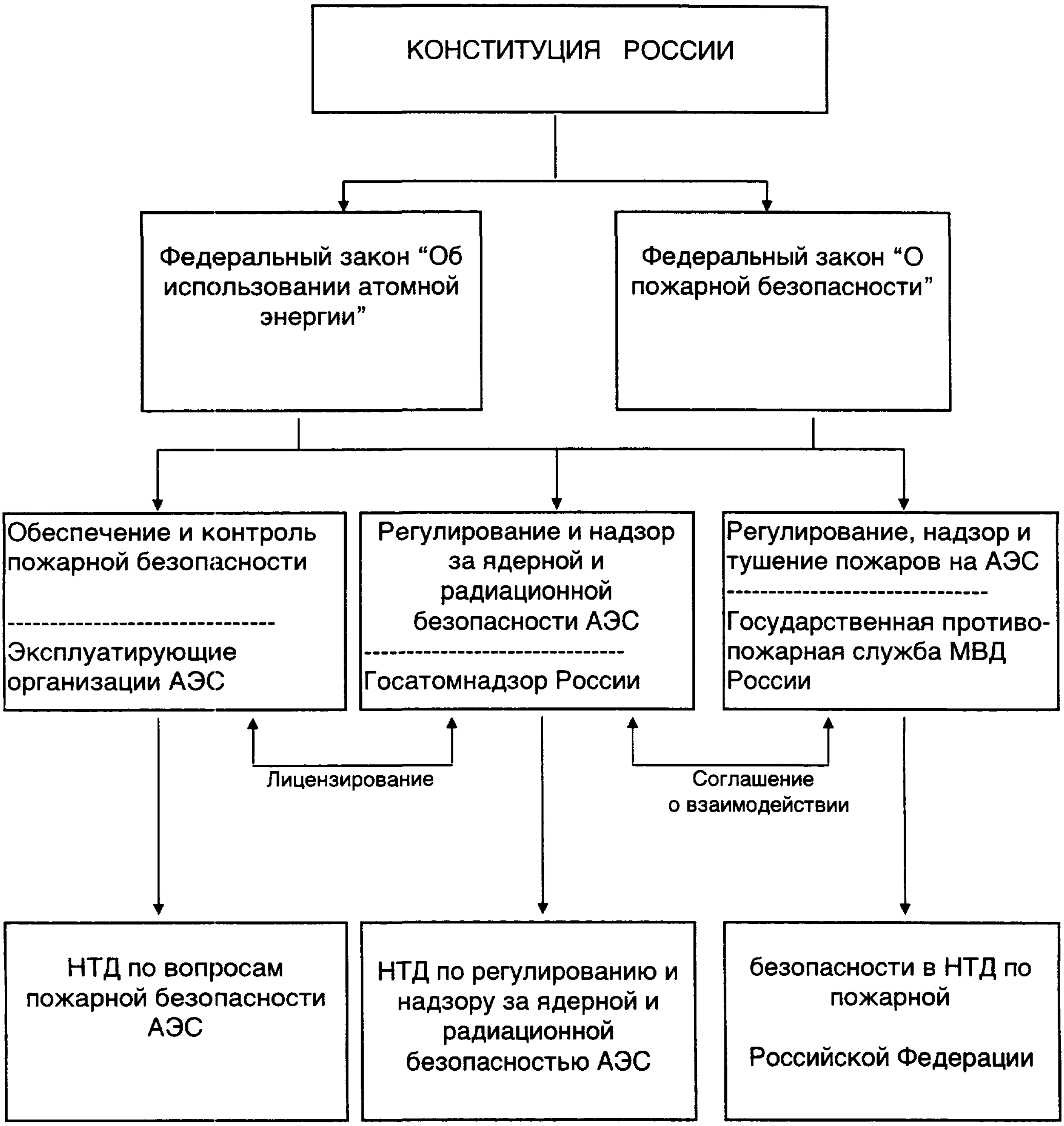
3. ЗАКЛЮЧЕНИЕ

Несмотря на существующие в Российской Федерации и атомной отрасли, в том числе, экономических трудностей государственными регулирующими органами и эксплуатирующими организациями Минатома России выполняется значительная работа по совершенствованию регулирующей деятельности и законодательной базы в области обеспечения ПБ АЭС.

Федеральный закон “Об использовании атомной энергии” поставил перед ГАН РФ дополнительные задачи по систематизации и развитию процедур, связанных с разрешительной деятельностью, совершенствованию нормативной базы, включая вопросы обеспечения ПБ на АЭС, что, по мнению ГАН РФ, должно положительным образом сказаться на повышении состояния безопасности АЭС России.

Приложение 1

Законодательная база и нормативно-правовое обеспечение в области регулирования пожарной безопасности АЭС России



Приложение 2

ПЕРЕЧЕНЬ НОРМАТИВНОЙ ДОКУМЕНТАЦИИ ГОСАТОМНАДЗОРА РОССИИ, ИСПОЛЬЗУЕМЫХ ПРИ НАДЗОРЕ И РЕГУЛИРОВАНИИ ВОПРОСОВ ПОЖАРНОЙ БЕЗОПАСНОСТИ АЭС

Инструкция по надзору за ядерной и радиационной безопасностью атомных станций (РД-04-18-95), Москва (1995).

Нормы проектирования сейсмостойких атомных станций (ПН АЭ Г-9-027-91), Москва (1991).

Общие положения обеспечения безопасности атомных станций, (ОПБ-88), Москва (1988).

Положение о порядке выдачи временных разрешений на эксплуатацию блоков АЭС (РД-04-01-93), Москва (1993).

Положение о порядке выдачи временных разрешений Госатомнадзора России на строительство блоков атомных станций гражданского назначения (РД-04-07-94), Москва (1994).

Положение о лицензировании деятельности в области использования атомной энергии, Москва (1997).

Положение о порядке выдачи временных разрешений Госатомнадзора России на строительство блоков атомных станций гражданского назначения (РД-04-07-94), Москва (1994).

Положение о порядке выдачи годичных разрешений Госатомнадзора России на эксплуатацию блоков атомных станций первого поколения, Москва, (1997).

Правила проектирования сейсмостойких атомных станций (ПН АЭ Г-05-006-87), Москва (1987).

Правила пожарной безопасности при эксплуатации атомных станций (ППБ-АС-95), Москва (1995).

Правила пожарной безопасности в Российской Федерации, Москва (1993).

Противопожарные нормы проектирования атомных станций (ВСН-87), Москва (1987).

Техническое содержание требований по обеспечению безопасности (ТС ТОБ АС-87), Москва (1987).

Требования к содержанию отчета по безопасности АЭС с реакторами типа ВВЭР и РБМК (ПН АЭ Г-01-036-95), Москва (1995).

Требования к эксплуатирующей организации атомной станции (РД-04-03-93), Москва (1993).

Федеральный закон об использовании атомной энергии, Москва (1995).

Федеральный закон о пожарной безопасности, Москва (1994).

Федеральный закон о радиационной безопасности населения, Москва (1995).

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ДЕЯТЕЛЬНОСТЬ УКРАИНСКОГО РЕГУЛИРУЮЩЕГО ОРГАНА ПО ОБЕСПЕЧЕНИЮ ПОЖАРНОЙ БЕЗОПАСНОСТИ ЭНЕРГОБЛОКОВ АЭС УКРАИНЫ

XA9847525

Г. ЛЯДЕНКО
Управление лицензирования ядерных установок,
Администрация ядерного регулирования,
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Украина

Abstract-Аннотация

THE UKRAINIAN REGULATORY AUTHORITY'S ACTIVITIES FOR ENSURING FIRE SAFETY AT UKRAINE'S NUCLEAR POWER PLANTS.

Ukraine's "Nuclear Energy Utilization and Radiation Safety Act", the basis for nuclear legislation in Ukraine, was passed in 1995. Pursuant to the Act, the Nuclear Regulatory Administration (NRA) of Ukraine's Ministry for Environmental Protection and Nuclear Safety is the main national body responsible for State safety regulation of the utilization of nuclear energy. The regulatory activities of the NRA cover — inter alia — those measures for the prevention of fires and for fire-fighting at nuclear facilities which affect nuclear safety, and the paper describes its regulatory activities in the fire safety field. The work of the Ukrainian regulatory authority responsible for ensuring fire safety at nuclear power plants consists in: (1) evaluating the designs of and analysing the fire safety arrangements at operating NPP units and verifying compliance with nuclear and radiation safety criteria in the event of fires at NPPs; (2) verifying that design changes made in the course of the modernization and upgrading of fire safety equipment at operating NPP units conform to the relevant safety standards and rules; (3) formulating basic rules for NPP fire safety which ensure nuclear and radiation safety; and (4) supervising the activities of operating organizations.

ДЕЯТЕЛЬНОСТЬ РЕГУЛИРУЮЩЕГО ОРГАНА УКРАИНЫ ПО ОБЕСПЕЧЕНИЮ ПОЖАРНОЙ БЕЗОПАСНОСТИ ЭНЕРГОБЛОКОВ АЭС УКРАИНЫ.

В 1995 году в Украине был принят закон "Об использовании ядерной энергии и радиационной безопасности", который является основополагающим в ядерном законодательстве Украины. Согласно этому закону основным национальным органом, который осуществляет государственное регулирование безопасности использования ядерной энергии, является Администрация ядерного регулирования Министерства охраны окружающей среды и ядерной безопасности Украины. Регулирующая деятельность охватывает те мероприятия по пожарной защите ядерных установок, которые влияют на ядерную безопасность. В докладе дается обзор деятельности регулирующего органа Украины в этой области. Работа регулирующего органа Украины по обеспечению пожарной безопасности АЭС ведется в следующих направлениях: (1) оценка проектов и анализ противопожарной защиты действующих энергоблоков, проверка обеспечения выполнения критериев ядерной и радиационной безопасности при пожаре на АЭС; (2) анализ соответствия нормам и правилам по безопасности изменений проектов АЭС, связанных с модернизацией и реконструкцией противопожарного оборудования действующих реакторных установок; (3) разработка нормативной базы по противопожарной защите АЭС, связанной с обеспечением ядерной и радиационной безопасности; (4) надзор за деятельностью эксплуатирующих организаций.

1. ВВЕДЕНИЕ

В 1995 году в Украине был принят Закон "Об использовании ядерной энергии и радиационной безопасности" /1/, который является основополагающим в ядерном законодательстве Украины.

Этот Закон устанавливает приоритет безопасности человека и окружающей среды, права и обязанности граждан в сфере использования ядерной энергии, регулирует деятельность, связанную с использованием ядерных установок и источников ионизирующего излучения, устанавливает правовые основы международных обязательств Украины, касающиеся использования ядерной энергии.

Согласно этому Закону национальным органом, который осуществляет государственное регулирование безопасности использования ядерной энергии, является Министерство охраны окружающей среды и ядерной безопасности Украины, а именно, Администрация ядерного регулирования.

2. ОБЗОР АЭС УКРАИНЫ

По количеству ядерных установок Украина занимает 7-е место в мире и 5-е в Европе. В Украине действуют 16 энергоблоков, размещенных на 5 площадках. Размещение АЭС по территории Украины показано на рис 1.

Основу реакторного парка Украины составляют водо-водяные реакторы типа ВВЭР-1000 (11шт.), реакторы ВВЭР-440 (2шт.) и уран-графитовые канальные реакторы типа РБМК-1000 (3шт.). Еще пять энергоблоков с реакторами типа ВВЭР-

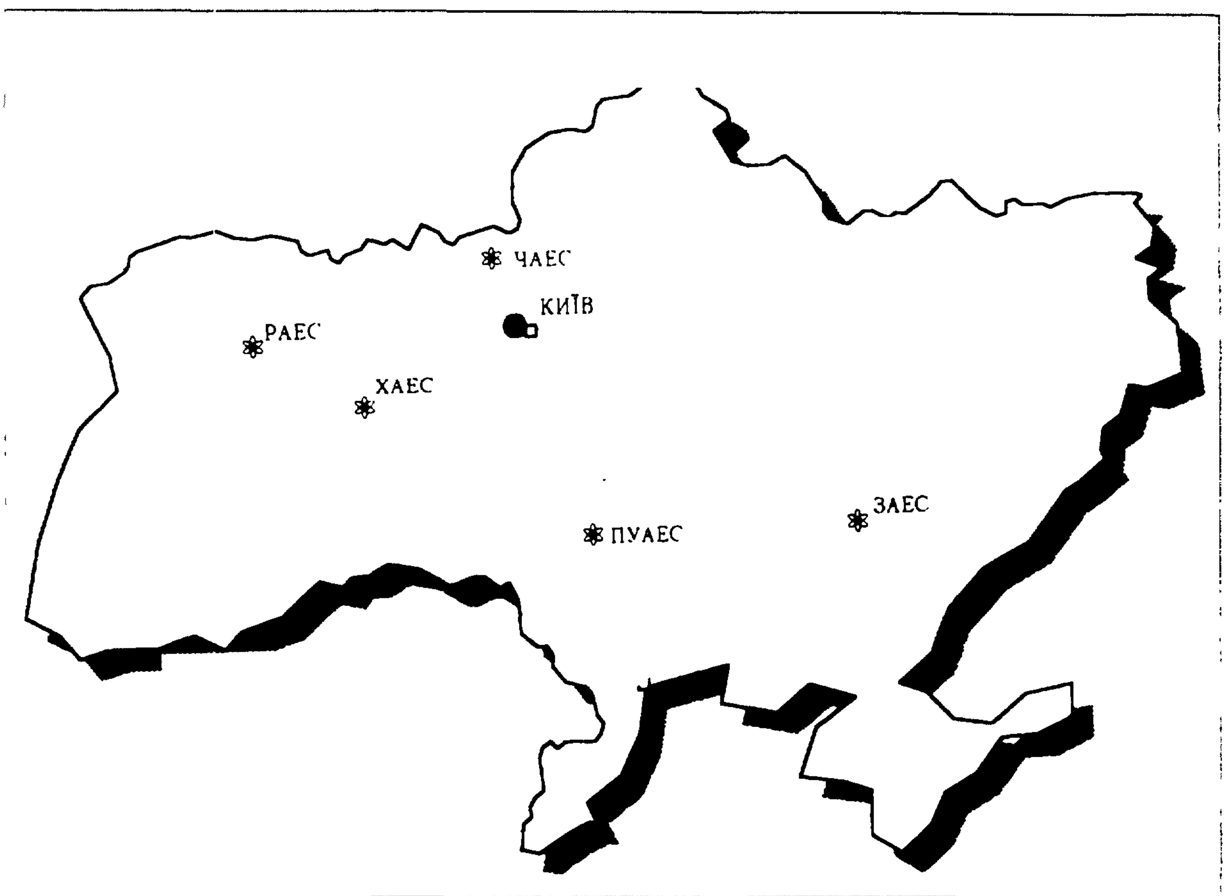

Рис.1

1000 находятся на стадии строительства на площадках ЮУ АЭС, Ровенской АЭС и Хмельницкой АЭС на разной ступени строительной готовности. Запорожская станция, с введением в действие энергоблока №6 в 1995 году после снятия моратория, стала самой мощной станцией Европы. Второй энергоблок ЧАЭС, начиная с 1996г., после пожара в турбинном отделении, законсервирован, блок №1 ЧАЭС окончательно остановлен в ноябре 1996года для проведения комплексных инженерных исследований, как преддверия деятельности по проектированию снятия с эксплуатации. Информация про АЭС Украины приведена в таблице.

3. ДЕЯТЕЛЬНОСТЬ РЕГУЛИРУЮЩЕГО ОРГАНА УКРАИНЫ

Все действующие АЭС Украины были спроектированы по нормам безопасности 70-х годов, действовавшим в бывшем СССР.

В начале своего становления регулирующий орган Украины в феврале 1992г. предложил администрациям АЭС осуществить комплексную переоценку безопасности действующих блоков (в том числе и пожарной безопасности).

Переоценка безопасности была необходима по следующим причинам:

- в 1986г в государстве произошла крупнейшая в истории ядерная авария (имеется в виду авария на ЧАЭС);
- в бывшем СССР были введены новые нормативные документы по безопасности;
- необходимость оценки влияния эксплуатационной практики;
- отсутствие в проектах АЭС таких важных разделов, как обеспечение качества, метрологическое обеспечение, влияние внутренних и внешних факторов на безопасность АЭС, анализ запроектных аварий, а также достаточные проработки вопросов снятия АЭС с эксплуатации;
- необходимость применения современных методов анализа различных действительных режимов эксплуатации;
- необходимость использования вероятностных методов для обоснования реализуемых на АЭС технических решений.

Переоценка безопасности действующих блоков АЭС Украины должна была преследовать такие цели:

1. Получение объективной оценки уровня безопасности в соответствии с современными национальными требованиями и рекомендациями международных организаций.
2. Выработка обоснованного комплекса мероприятий с целью повышения безопасности действующих блоков и реализация соответствующих решений на строящихся блоках до ввода их в эксплуатацию.

Все действующие энергоблоки АЭС в соответствии с Законом Украины "Об использовании ядерной энергии и радиационной безопасности" до 2000 года должны получить лицензии регулирующего органа на постоянную эксплуатацию. Учитывая, что в бывшем Советском Союзе не выполнялся полномасштабный анализ безопасности АЭС, необходимым условием получения лицензии на постоянную эксплуатацию является представление в регулирующий орган отчетов по анализу безопасности.

Пожарная безопасность на АЭС Украины регулируется, как и во многих странах, двумя регулирующими органами:

В соответствии с Законом Украины "Про пожарную безопасность" /2/ Министерство внутренних дел Украины осуществляет регулирование и надзор за обеспече-нием пожарной безопасности на АЭС, как объекте общепромышленного назначения. Целью регулирования и надзора при этом является обеспечение безопасности чело-века при пожаре и сохранность материальных ценностей.

Министерство охраны окружающей среды и ядерной безопасности Украины осуществляет регулирование и надзор за обеспечением пожарной безопасности на АЭС, как объекте атомной энергетики. Целью при этом является обеспечение ядерной и радиационной безопасности во время и после пожара на АЭС. Мы рассматриваем те мероприятия по противопожарной защите АЭС, которые влияют на ядерную и радиационную безопасность.

4. ОСНОВНЫЕ НАПРАВЛЕНИЯ РАБОТЫ РЕГУЛИРУЮЩЕГО ОРГАНА УКРАИНЫ

Работа регулирующего органа по обеспечению пожарной безопасности АЭС ведется в таких направлениях:

- оценка проектов и анализ противопожарной защиты действующих энергоблоков, проверка обеспечения выполнения критериев ядерной и радиационной безопасности при пожаре на АЭС;
- анализ соответствия нормам и правилам по безопасности изменений проектов АЭС, связанных с модернизацией и реконструкцией противопожарного оборудования действующих реакторных установок;
- надзор за деятельностью эксплуатирующих организаций;
- разработка нормативной базы по противопожарной защите АЭС, связанной с обеспечением ядерной и радиационной безопасности.

Согласно современным требованиям нормативных документов, оценка проектов и анализ достаточности мероприятий противопожарной защиты АЭС должны проводиться на основе результатов выполнения анализа пожарного риска. Полный количественный анализ пожарного риска, соответствующий рекомендациям МАГАТЭ /3//4/, не выполнен до сих пор ни на одной станции Украины.

Правда, в этом году в рамках программы TACIS/PHARE на Ровенской АЭС (блоки №1, 2) с помощью ЭДФ проводились работы по анализу пожарной уязвимости помещений дизель-генераторной станции с целью улучшения пожарной защиты этого здания, на основании которого разработан перечень мероприятий.

Анализ пожаров, имевших место на блоке №2 Игналинской АЭС (1988г.) и на блоке №2 Чернобыльской АЭС (1991г.) и других АЭС, показал, что основными причинами пожаров было несоответствие пожарной безопасности кабельного хозяйства АЭС требованиям норм и правил по безопасности. Поэтому, учитывая то, что переоценка безопасности АЭС, подготовка отчета по анализу безопасности (с проведением и анализа пожарного риска) - это длительный процесс, регулирующий орган Украины предпринял срочные меры по повышению пожарной безопасности кабельного хозяйства АЭС Украины.

В 1994 году Коллегия Госатомнадзора Украины (так назывался тогда регулирующий орган Украины) приняла Решение от 16.11.94г. №26 “О пожарной безопасности АЭС Украины”/5/, в котором обязала эксплуатирующие организации принять конкретные меры по обеспечению пожарной безопасности кабельного хозяйства АЭС. Во исполнение требований регулирующего органа орган государственного управления в сфере использования атомной энергии (так называемый Госкоматом) разработал “Комплексную целевую программу обеспечения пожарной безопасности кабельных сооружений АЭС Украины на 1995-1996 год”/6/.

Этой программой было предусмотрено выполнение следующих мероприятий:

- разработка отраслевых нормативных документов по пожарной безопасности ка-бельного хозяйства, в том числе и правил пожарной безопасности для АЭС;

- проверка проектов, а также фактического состояния энергоблоков АЭС на соответствие действующим нормам и правилам по безопасности, выявление отклонений от требований НТД;

- разработка условий безопасной эксплуатации кабельного хозяйства;

- разработка и выполнение программы приведения кабельного хозяйства энерго-блоков в соответствие с требованиями действующих норм и правил по безопасности.

Эти работы включали в себя:

- проверку огнестойкости противопожарных барьеров, ограничивающих различные пожарные зоны, в которых размещаются разные каналы систем безопасности, с проведением огневых испытаний всех узлов и элементов кабельного хозяйства, вклю-чая противопожарные двери, проходки, огнезадерживающие клапаны и кабели, покрытые огнезащитным покрытием;

- при необходимости покрытие кабелей огнезащитными материалами, разрешен-ными к применению регулирующими органами;

- выполнение анализа влияния огнезащитных покрытий на допустимые токовые нагрузки;

- определение термической и пожарной стойкости кабелей и разработка мероприятий по предотвращению самопроизвольных включений механизмов в результате коротких замыканий между жилами кабелей;

- для помещений, в которых отсутствует физическое разделение разных каналов систем безопасности, проведение анализа обеспечения выполнения критериев безо-пасности для ядерной установки (возможность безопасного останова реактора и его поддержание в подкритическом состоянии во время и после пожара, длительный отвод остаточных тепловыделений после останова реактора и непревышение допусти-мых проектом пределов выбросов радиоактивных веществ);

- реализация организационных мероприятий по обеспечению пожарной безопасности.

В дополнение к этой “Комплексной программе...” регулирующий орган выпустил информационное письмо АЯР №9-96 /7/, где уточнил требования к проведению анализа безопасности кабельного хозяйства.

Станции должны:

- провести обследование кабельного хозяйства каждого энергоблока и выявить отступления от действующих норм и правил;
- проанализировать влияние выявленных отступлений от требований НТД на функционирование систем безопасности и, следовательно, на безопасность АЭС;
- разработать мероприятия по предупреждению отказов и/или уменьшению последствий пожара (условия безопасной эксплуатации кабельного хозяйства);
- определить на случай возникновения пожара в конкретном помещении допустимость и продолжительность работы реакторной установки в основных состояниях и необходимость перевода ее в другое состояние;
- представить в регулирующий орган документы: “Условия выполнения вышеперечисленных критериев безопасности” и “Программу приведения кабельного хозяйства АЭС в соответствие требованиям действующих НТД.

Выполнение этих требований введено в особые условия лицензии на эксплуатацию энергоблоков.

Выполнение требований регулирующего органа является первым этапом работ по выполнению анализа опасности возникновения пожара на АЭС.

Конечно, мы понимаем, что такой анализ без проведения полноценного количественного анализа несовершенен и носит качественный характер, но на сегодняшний день даже такой анализ принесет несомненную пользу для обеспечения безопасности АЭС.

Работа по выполнению полного количественного анализа опасности возникновения пожара с применением рекомендаций международных организаций также намечается в перспективе.

Мы принимаем участие в проекте, финансируемом Департаментом энергетики США, по внедрению на Украине разработанной ими специально для проектов реакторов ВВЭР и РБМК бывшего Советского Союза "Методология оценки защиты от огня активной зоны реакторов РБМК и ВВЭР" ("Reactor Core Protection Evaluation Methodology for Fires at RBMK and VVER Nuclear Power Plants")/8/.

Для Запорожской АЭС будет разработан пилотный проект и проведена работа по адаптации этой "Методологии..." для украинских реакторов. Работы начнутся уже в этом году.

Кроме того, по программе двухстороннего сотрудничества с Обществом по безопасности станций и реакторов (GRS) Федерального Министерства окружающей среды, защиты природы и безопасности реакторов (BMU) Германии, мы предполагаем ознакомиться с их методикой проведения анализа опасности возникновения пожара, и также провести такой анализ на одной из АЭС Украины.

После проведения этих работ регулирующий орган выпустит рекомендации по проведению анализа опасности возникновения пожара для украинских АЭС.

Таблица 1

Блок АЭС	Тип реактора	Мощность, МВт (ел.)	Дата начала эксплуатации	Время эксплуатации, на 01.01.97р. (лет/мес.)
1	2	3	4	5
	Запорожская	АЭС		
1.	ВВЭР-1000/В-320	1000	10.10.84	13/1
2.	ВВЭР-1000/В-320	1000	02.07.85	12/4
3.	ВВЭР-1000/В-320	1000	10.12.86	10/11
4.	ВВЭР-1000/В-320	1000	24.12.87	9/10
5.	ВВЭР-1000/В-320	1000	31.08.89	8/3
6.	ВВЭР-1000/В-320	1000	19.10.95	2/1
	Южно-	Украинская	АЭС	
1.	ВВЭР-1000/В-302	1000	22.12.82	14/11
2.	ВВЭР-1000/В-338	1000	06.01.85	12/10
3.	ВВЭР-1000/В-320	1000	20.09.89	8/2
4.	ВВЭР-1000/В-320		в стадии	строительства
	Ровенская	АЭС		
1.	ВВЭР-440/В-213	440	31.12.80	16/11
2.	ВВЭР-440/В-213	440	30.12.81	15/11
3.	ВВЭР-1000/В-320	1000	24.12.86	10/6
4.	ВВЭР-1000/В-320	1000	в стадии	строительства
	Хмельницкая	АЭС		
1.	ВВЭР-1000/В-320	1000	31.12.87	9/11
2.	ВВЭР-1000/В-320	1000	в стадии	строительства
3.	ВВЭР-1000/В-320	1000	в стадии	строительства
4.	ВВЭР-1000/В-320	1000	в стадии	строительства
	Чернобыльская	АЭС		
1.*	РБМК-1000	1000	26.09.77	19/2
2.**	РБМК-1000	1000	21.12.78	13/8
3.	РБМК-1000	1000	10.11.81	15/2

* остановлен в декабре 1996г. для последующего снятия с эксплуатации;
**остановлен после пожара 11.10.91г.

- **Следующее направление деятельности регулирующего органа** - анализ соответствия противопожарной защиты изменений проектов АЭС, связанных с модернизацией и реконструкцией действующих реакторных установок, нормам и правилам по безопасности;

Для повышения безопасности (в том числе и пожарной безопасности) действующих энергоблоков разработаны и выполняются различные программы модернизации АЭС /9/-/11/.

Учитывая то, что глубокий анализ опасности возникновения пожара на блоках АЭС не проводился, и обосновать безопасность энергоблока при наличии отклонений от действующей НТД эксплуатирующая организация не может, поэтому программы повышения пожарной безопасности АЭС построены, как правило, на ликвидации отступлений от требований действующей НТД. При этом, учитывается опыт эксплуатации аналогичных проектов АЭС и рекомендации международных организаций.

Программы модернизации включают такие вопросы пожарной защиты и защиты от взрывов:

- замена противопожарных дверей на двери с огнестойкостью 1,5 часа в помещениях, содержащих системы, важные для безопасности;
- покрытие кабелей и других элементов огнезащитными материалами;
- замена пластикатного покрытия полов на огнестойкие;
- введение противопожарных барьеров между разными каналами систем безопа-сности;
- повышение огнестойкости металлоконструкций каркаса машзала;
- установка огнезадерживающих клапанов в воздуховодах систем вентиляции;
- внедрение систем газового пожаротушения для тушения помещений, содержащих электрическое и электронное оборудование систем контроля и управления;
- модернизация систем пожарной сигнализации (имеющаяся система пожарной сигнализации не аттестована на применение в условиях АЭС, сейсмичность и радиационная стойкость не отвечает необходимым требованиям);
- повышение надежности водоснабжения систем водяного пожаротушения;
- разделение больших пожарных отсеков на меньшие;
- повышение термической и пожарной устойчивости электрических присоединений;
- обеспечение контроля концентрации водорода в гермооболочке;
- внедрение схемы автоматического по сигналу "Пожар" сброса водорода из кор-пуса генератора за пределы машзала;
- замена горючих смазочных масел в системе смазки главных циркуляционных насосов на негорючие;

и т.д.

К сожалению, необходимо отметить, что часто выполнение этих мероприятий срывается из-за недостатка финансирования.

Другим направлением деятельности регулирующего органа является разработка нормативной базы по обеспечению пожарной безопасности энергоблоков АЭС, связанной с обеспечением ядерной и радиационной безопасности.

Подготовлены такие документы:

- "Требования к противопожарной защите энергоблоков АЭС, связанные с обеспечением ядерной и радиационной безопасности" /12/. Первая редакция этого нормативного документа уже разработана и находится на обсуждении. Готовится 2-ая редакция документа. В этом нормативном документе изложены основные критерии и принципы обеспечения ядерной и радиационной безопасности во время и после пожара на АЭС.
- "Процедура инспекции по проверке безопасного останова реактора при пожаре на энергоблоках АЭС"/13/. Находится на рассмотрении.

Кроме этого, ведется разработка отраслевой нормативной базы с нашим согласованием:

- проведен пересмотр нормативного документа бывшего Советского Союза ВСН01-87 "Противопожарные нормы проектирования атомных станций с водоохлаждаемыми реакторами на тепловых нейтронах" /14/, первая редакция документа находится на стадии согласования;
- выпущена первая редакция отраслевого документа "Правила пожарной безопасности АЭС" /15/, находится на согласовании;
- разрабатываются отраслевые общие технические требования для ОЗС, применяемых на АЭС /16/;
- выпущен отраслевой документ "Расчетная методика "Определение коэффициентов снижения длительно допустимых токовых нагрузок силовых кабелей при покрытии их ОЗС и проходе через огнестойкие заделки и огнепреградительные пояса"/17/.

Надзор за деятельностью АЭС

На всех этапах жизненного цикла АЭС инспекциями регулирующих органов осуществляется надзор за деятельностью АЭС. При этом:

- проводится надзор за соответствием состояния АЭС и режимов эксплуатации требованиям нормативно-технической документации и условиям выданной лицензии,

- проверяется соответствие пожарной безопасности АЭС нормам и правилам по безопасности,

- проверяется возможность сохранения безопасности АЭС, защиты персонала при пожаре и после пожара,

- при наличии отклонений от требований НТД проверяется наличие компенсирующих мероприятий и оценивается достаточность их для обеспечения безопасности АЭС.

ВЫВОДЫ:

1. Установлены различные недостатки в обеспечении пожарной безопасности АЭС самими эксплуатирующими организациями, в том числе и с помощью международных экспертов. Проводятся определенные мероприятия с целью повышения пожарной безопасности АЭС.
На сегодняшний день основные проблемы для Украины:
-финансирование проектов модернизации старых и строящихся энергоблоков АЭС;
-необходима помощь при проведении анализа опасности возникновения пожара действующих энергоблоков.

2. Регулирующий орган Украины прикладывает определенные усилия в обеспечении пожарной безопасности на АЭС Украины, доведении пожарной безопасности действующих энергоблоков до уровня современных норм и правил по безопасности.

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GENERAL FIRE PROTECTION GUIDELINES FOR EGYPTIAN NUCLEAR INSTALLATIONS

XA9847526

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Abstract

The purpose of this paper is to establish the regulatory requirements that will provide and ensure fire protection of Egyptian nuclear installations. Two or more classes of occupancy are considered to occur in the same building or structure. Fire protection measures and systems were reviewed for four of the Egyptian nuclear installations. These are Egypt's first research reactor (ET-RR-1) building and systems, hot laboratories buildings and facilities, the building including the AECL type JS-6500 industrial cobalt-60 gamma irradiator "Egypt's Mega Gamma I" and Egypt's second research multi-purpose reactor (MPR). A brief review is given about fire incidents in Egypt, and descriptions of the only fire reported at one of the Egyptian nuclear installations over more than 35 years of operating these installations. The study outlines the various aspects of fire protection with a view to define the relevant highlights and scope of an Egyptian guidelines.

1. FIRE EVENTS IN EGYPT

Fire risk assessment studies indicate that the contribution of fires to severe accidents at large facilities is significant and that fire may contribute in excess of 50% of the total frequency of accidents leading to large scale losses. In addition to the public health risk represented by fires, the financial risk due to direct fire damage, as well as loss of revenues as a result of fire can be significant. In the last few years there were numerous fires that occurred in a number of industrial and petroleum facilities in Egypt and the evaluation of losses in many of them exceeded several millions Egyptian pounds (L.E.).

Some recent studies in the area of fire protection have been made in Egypt [1,2]. It became clear during data collection that there is a lack of statistics on fire incidents, research and suitably applicable codes.

The study and analysis of fire events in Egypt [1] had throw light upon the importance of assessing fire hazards and helped deeply the understanding of fire protection concepts and requirements.

1.1. Population, and Population Activities

One of the objectives of the study of fire events in Egypt was to investigate any relation that might exist between the frequency of fire incidents and the population of the district, the density of population, the activities of the inhabitants (industrial, agricultural and/or commercial) and other features.

Egypt's population nearly doubled from 9.7 million to over 19 million persons in the 50 years from 1897 to 1947. The next doubling took less than 30 years (from 1947 to 1976). In 1996, the Egyptian population was 61 million. The population growth rate, which was 1.5 percent annually at the beginning of this century, fell for a period and then began rising rapidly from the early 1950s, reaching a rate of approximately 2.5 percent in the early 1960s. For the period 1960-1967, the growth rate slackened, but by early 1980s it had risen again to nearly 3 percent [1].

The problems and risks associated with the rapid rate of population growth are complicated further by one basic fact about Egypt, namely the extreme scarcity of cultivable land relative to people. Over 97 percent of Egypt's 1996 population of 61 million is crowded on to less than four percent of the total area of one million square kilometers. The remaining 96 percent of the land area is desert.

The concentration of the population in the Nile Valley and the Delta, gives Egypt in 1996, density rates of 61 persons per square kilometer for the total area but over 1520 persons per square kilometers of inhabitable land. In Cairo, density reaches 28332 persons per square kilometer, now about 42% of the total urban population lives in Cairo and Alexandria.

Egypt has made great progress in the field of traditional industries, namely, spinning and weaving industries. Significant progress has also been made in modern industries, such as metallurgical and chemical and other engineering industries. The state contributed to a most efficient drive towards increasing production in many industries, such as iron and steel, ceramics and porcelain, cement, paper and fertilizer and the petroleum industry. Im Egypt about 33% of the population works in agriculture and 22% in the different industries.

1.2. Review and Analysis of Fire Events

Review and analysis of fire incidents in Egypt during twelve years, from Jan. 1980 to Dec. 1991 was performed [1]. This study is considered as the first comprehensive analysis of fire incidents in Egypt, the data required to analyze the fire incidents in Egypt was obtained from the yearly reports issued by the civil defense and general security organizations of the Ministry of Interior. During the phase of data collection it is obvious that there are incomplete records about fires and the way of reporting them. It is clear that fire protection, fire fighting, fire reporting systems and fire codes need to be reviewed.

The analyses include the distribution of fire events and their losses over the different governorates and over the months of the year, the primary causes, the classification of events according to the place of occurrence. Also the study throws some light on arson fire events. The study provides general picture of fire experience over time, for a greater insight more elements have be provided by the Ministry of Interior to describe fire incidents.

A summary of fire statistics for the total incidents reported in Egypt (1980 - 1991) is given in Table I. The recorded number of civilian deaths, civilian injuries and monetary losses, are given. The average annual number of fires was 21 thousands events (TFI) in the Republic resulted in 227 deaths and 752 civilian injuries in the average. The average annual value of the monetary losses due to fires was about eleven million L.E.

From the study the following rated were calculated: One injury for each 27 TFI and one death for each 92 TFI during the study period. The average values were: one TFI for every 2677 Egyptian person in the year and one TFI occurred in the year for each 48.2 km^2 of the republic area compared to 1.93 TFI/km^2 of the inhabitable land.

The distribution of incidents over the different governorates is more or less similar in all years of investigations where Cairo, Alexandria, Dakahliya and Giza governorates have the large numbers of fire incidents. These governorates are the ones with larger population and population intensity and they are industrial areas (more than 70% of the total industries are in Cairo and Alexandria).

Among all industrial facilities incidents those that had been occurred at petroluem companies were the most severe from the point of view of monetary losses. The incidents that had been occurred at village houses are the most severe regarding the No. of civilian deaths and injuries recorded in one incident (more than 110 injuries and 60 deaths in an incident).

TABLE I. TOTAL FIRE STATISTICS

Year	*Total Fires*	*Civilian Injuries*	*Civilian Deaths*	*Monetary Losses (10^3 L.E.)*
1980	15713	980	291	3586
1981	17185	663	179	12239
1982	17660	940	165	23310
1983	19476	761	310	8900
1984	20868	1029	269	7200
1985	21687	753	184	17043
1986	22773	657	303	8090
1987	24177	808	190	6699
1988	22776	806	198	9465
1989	23351	781	246	9139
1990	22417	808	182	16910
1991	21163	701	200	7045

About 110 arson fire felonies, on the average, was reported annually in Egypt, 50% of them occurred in Cairo, Alexandria and Menofia governorates (40% of the TFI were recorded in these three governorates) with 25% of them occurred in Cairo governorate.

The distribution of arson fires, according to places of their occurrence was as follows: The accidents occurred at shops, residences, coffee shops, farms, factories, schools, streets, mosques and other places From the study it has been found that about 50% of the arson fires occurred inside residential buildings followed by commercial shops and farms and annually about 4 arson felonies were recorded at schools and one or two at mosques.

Based on the results of the study and investigations done codes and standards need to be reviewed and applied. Also a comprehensive national fire incident reporting system have to be established to collect, analyze and disseminate information on fire events in order to be able to reduce their consequences. The main recommendations of the study include as the Ministry of Interior is the only organization in Egypt with overall responsibility for fire protection and fire fighting measures, it is recommended to have:

- Center for fire research whose scope extends from exploratory research on combustion to the development of computer programs to solve practical fire protection engineering problems.
- A bachelor degree in fire protection engineering has to be offered by one of the departments of a faculty of engineering.
- Fire protection basics have to be included as one of the educational subjects in preparatory and secondary schools, to raise the level of safety culture regarding fire protection issues.

2. EGYPTIAN NUCLEAR INSTALLATIONS

The Egyptian Atomic Energy Authority (AEA) has 4 research centers, the activities of the AEA run into four major fields: research and technological projects, radiation protection and safety, society services activities, regional and international cooperation. The AEA houses some major and research facilities, among them, Egypt's first Research Reactor ET-RR-1, the Hot laboratories Center at Inches, and the Industrial Irradiator Facility (Egypt's Mega Gamma-1) at Nasr City.

The major near commissioning projects are Egypt's Second Research (Multi - Purpose) Reactor (MPR) and the Cyclotron Accelerator.

The MPR is an advanced swimming pool research reactor contracted with INVAP Argentina in Sept. 1991. This is 22 MWT reactor will be used in radiostope production, materials testing research, reactor physics research and reactor thermal engineering research.

The Cyclotron complex is based on a compact AVF cyclotron of Russian type MGC-20 with K = 20. The accelerator is intended to be used in a multidsciplinary way. The complex contains provisions for radioisotope production, fast neutron research applications, use of cyclotron beams in nuclear analytical techniques, biomedical and nuclear medicine applications, and surface modifications and treatment. Shielding design of walls, floors and ceilings are done in accordance with ICRP-60 recommendations with dose limit rates to non-occupational exposed individuals not exceeding 0.5 mSv/y.

Both the reactor and the cyclotron are to be commissioned by the end of 1997. The fire protection system design for Egypt's second research reactor MPR was reviewed by the staff of the Egyptian regulatory body. A brief description of the system is given below.

The fire alert system is responsible for detecting and providing early fire alarm inside the reactor building. The system is controlled by a fire alert station located in the security guards office, where security personnel are presented at all times. An alarm panelboard is installed in the Control Room. The reactor building has been divided into 13 fire sectors or zones. This division has been performed according to the building's physical characteristics and its operational areas. Also, Each floor is divided into two fire sectors, one to cover the cold zone and the other to cover the hot or controlled zone.

Fire alarm center operates with conventional type fire alarms, extinction system automatic tripping happens when activation of two detectors, from the circuit of room to be protected is detected. In control room and in auxiliary control room fire alarm repetition panels are installed. Electronic thermal detectors, ionic smoke detectors, and photoelectric smoke detectors are used and distributed along the different fire sectors considering the fire risk of each area.

The fire extinguishing system consists of the following elements: fire protection water network which will include a perimetral external ring and a line of internal hydrants, one per floor, located in the building's cold zones, automatic extinction installation by, means of a gaseous agent in certain rooms (control room, auxiliary control room, emergency control room, electrical board) and set of manual extinguishers of various types and capacities appropriately distributed among cold and hot zone rooms and premises.

3. FIRE AT THE INDUSTRIAL COBALT-60 GAMMA IRRADIATOR "EGYPT'S MEGA GAMMA I "

The Egyptian Atomic Energy Authority possesses and operates AECL type JS-6500 industrial Cobalt-60 gamma irradiator "Egypt's Mega Gamma I". The facility has been in operation at the National Centure for Radiation Research and Technology (NCRRT) since January 1979 at an initial Cobalt-60 capacity of 400 kCi.

The plant is furnished with a ventilated concrete biological shield, a water pool for source storage, a principal mechanical conveyor, an extra research channel for pilot irradiation of high density products, a source passes mechanism with peneumatic pushers and all other devices for interlocks, radiation safety and absorbed dose measurements.

On September 5, 1995 a fire started inside the irradiation rom. The alarm system was actuated at 8:23 in the morning, anti-fire measures were taken and the fire was totally extinguished at 11 : 30. The fire was initially detected automatically by overhead's signal from the control panel. The source was automatically lowered to its safe position in the pool.

At the time of fire occurrence, the plant was in operation since four days. The plant was operated using the research channel. The main conveyor was not used and it was loaded by 60 dummy carboard boxes eatch of 30 kg. The research channel was loaded by eye ointment aluminum tubes of tetra cyclone packed in small (10 x 10 x 10 cm) carboard boxes. The ointment was irradiated in an experiment for Pfizer Drug Company for a dose of 2 MRad.

More than 2 tons of dummy boxes were destroyed. There was not any releases of radioactivity as a result of the fire. Also no one was injured due to fire or by any other cause. Outside fire brigade arrived in few minutes after their call (after about 10 minutes from the detection of the fire). They started to extinguish the fire after being sure that there was not any radiation releases. Water alone was first used for about 90 minutes and then they used water-chemical foam to fill up the whole irradiation room.

The cause of fire was investigated by a group of specialists from the Criminal Lab and experts of NCRRT. They came to a conclusion that the fire started at the bacalite base of neon fluorescent bulb as a result of bad contact. After fire occurrence all the systems of the facility had been reviewed and lighting and fire protection systems are updated. This fire is the only reported fire at any one of the Egyptian nuclear installations over about 35 years of operation.

4. SPECIAL PROBLEMS IN FIRE PROTECTION AT NUCLEAR INSTALLATIONS

The special problems encountered in nuclear installations are due to the presence of different radioactive substances (solid, liquid or gaseous) with varying activity and decay rates. These radioactive substances are the cause for the division of the installation premises into what are sometimes called controlled and uncontrolled areas in order to limit the exposure of the employees when working in different parts of the installation. The division of the premises into controlled and supervised areas affects the arrangement of the fire protection systems and the design of ventilation systems.

The presence of gaseous or airborne radioactive substances requires pressure differentials to maintain the flow of air from the less towards the more contaminated rooms and regions. Smoke extraction methods must accommodate this principle.

Manual fire fighting in a nuclear installation may prove to be difficult and time consuming operation, since the fire fighters must be given sufficient protection against radiation exposure, be it by limiting the exposure time or by wearing special protective equipment. The radiation dose limits to which fire fighters and other emergency personnel may expose themselves is a subject on which no simple statement can be made. One kind of exposure comes from external exposure to ionizing radiation. Another comes from radioactive substances which may be inhaled or ingested. The exact limits should be defined by the emergency program established for the particular installation.

5. FIRE PROTECTION GUIDELINES

The National Center for Nuclear Safety and Radiation Control (NCNSRC) is the national authority responsible for the surveillance of the safety of nuclear facilities. The regulatory activities encompass also the fire protection of the facilities in so far as they affect the nuclear safety of the facilities. In its regulatory work, NCNSRC takes into account the activities of other authorities and organizations.

General fire protection guidelines which have to be enforced are described in this paper after reviewing the conditions at Egypt's nuclear installations and after going through the available guides and standards [3-9].

The requirements given in this paper are not applied at the exisiting Egyptian nuclear facilities. The fire protection modifications that may be needed for them are considered case by case.

5.1. Design Requirements

Protection from fire and fire related explosions assumes importance in the overall design of a nuclear facility in so far as it forms a crucial part of the safety considerations. Therefore planning for fire protection shall be an integral part of the design stage and not an afterthought. Fire protection shall continue to be a well planned and implemented program throughout the life of the facility.

5.1.1. Structural fire protection

Structural fire protection measures shall be capable of, as far as possible, ensuring alone the safety of a nuclear facility in the event of fire. Therefore, the functional design and layout considerations of a nuclear facility and its buildings form the prerequisites for adequate fire prevention and protection. One design aspect shall be the housing of the proportions of the facility most important to nuclear safety in separate buildings apart from the conventional parts of the facility, there by facilitating protection against fire of the items important to nuclear safety. The buildings containing items important to nuclear safety should be fire resistant.

The electrical power supplies between the facility and the national grid shall be arranged in such a way that the probability of losing all supplies simultaneously due to a fire is minimized.

The process electrical and instrumentation systems at the facility shall be diversified and partitioned into different fire areas in such a way that in case one fire area is destroyed, there are still sufficient number of systems available to ensure the safety of the facility. The boundary between the controlled and uncontrolled zones shall also be the boundary between fire areas.

The requirements set forth in the safeguards shall also be taken into account in the design and dimensioning of the fire areas, access and escape routes and fire doors.

The fire areas should have a minimum fire resistance of two hours. The access and escape routes needed for the safe shutdown of the facility, the access routes for fire brigades, at least one emergency exit in each building, shall be so designed and constructed that these areas can be used safely at least for two hours under postulated fire conditions [4].

The fire resistance of the separating elements of fire boundaries such as doors and hatches, cable and pipe penetrations shall be equal to that required for the walls, the floors, and the ceiling structures.

Ventilation should not degrade fire protection of the facility; fire areas containing redundant systems important to nuclear safety should not be provided with mutual ventilation systems that could increase fire hazards. In the design of the ventilation systems, it shall be taken into consideration that in the event of a fire they can quick switched off quickly and reliably.

5.1.2. Active fire protection

The objective of active fire protection measures is an early detection and effectively extinguishing a fire. Acive fire protection comprises a fire detection and alarm system, and fire extinguishing systems and other fire suppression arrangements as complementary measures to structural fire protection. However, the safety of the facility shall not, in any parts, be dependent on the active fire protection measures alone.

To facilitate fast suppression of fire and to minimize damage and hazards, effective fire extinguishing systems should be designed for the facility. Fixed reliable fire extinguishing systems shall be designed for the facility. They should be provided for the following rooms and systems, irrespective of the lay-out design of the facility:

(1) Cable spaces containing redundant cables which are important to safety not housed in separate fire areas.
(2) Large oil systems for the main circulation pumps.
(3) Diesel generators (if existed)
(4) Spaces and systems for which considerable amounts of radioactive substances can be released into rooms or into the environment by a fire.

Removal of extinguishing water shall be arranged from rooms equipped with fixed water extinguishing systems. The buildings of the factility should be provided with adequate access and escape routes. These routes shall be spacious and easy to pass through.

The facility shall be equipped with an emergency lighting with the purpose of ensuring safety of passage inside the building when the normal lighting is out of order. Emergency lighting means both signal lights and standby lighting. The purpose of standby lighting is to remain on or to be switched on automatically or manually when the normal lighting goes out. Emergency lighting shall also be located near the signs indicating the escape routes and emergency exists.

5.2. Construction Permit

Before the construction permit, the preliminary Safety Analysis Report (PSAR), and the complementary topical reports must be submitted to the regulatory body. These documents shall provide a description of how the fire protection requirements are met in the design of the facility. The division of items should be as follows [3]:

(1) Description of regulations, guides and standards used in the design.
(2) Description of fire loads.
(3) Description of fire areas.
(4) Description of ventilation in the event of a fire.
(5) Preliminary description of the fire detection and alarm system, and fire extinguishing systems.
(6) Description of the escape routes and emergency exists.
(7) Description of the fire hazards analyses.

5.3. Fire Hazard Analysis

The purposes of the fire hazard analysis are:(1) To identify items important to safety

(2) To analyze the anticipated fire growth and the consequences of the fire with respect to items important to safety.
(3) To determine the required fire resistance of fire barriers.
(4) To determine the type of fire detection and protection means to be provided.
(5) To identify cases where additional fire separation or fire protection is required, especially for common mode failures, in order to ensure that items important to safety will remain functional during and following a credible fire.
(6) To verify that the safety systems required to shut the facility down, remove residual heat (if required), and contain radioactive material are protected. They should be protected against the consequences of fires so that they are still capable of performing their safety functions.

To secure effective nuclear safety for the installation, the fire hazard analysis must cover all areas of the site, including the non-nuclear facilities. Assessment of all the site areas is necessary to ensure that all the fire hazards which potentially threaten nuclear safety have been addressed. Fire protection measures (both passive and active) should also be considered for any area containing concentrations of combustibles, even though the area may not contain or expose nuclear safety system. This additional protection may be provided in order to minimize both property damage and installation down that could occur as a result of fire.

For identification of fire hazards and safety systems, the information that must be obtained can be separated into seven categories: fire compartment inventory, combustibles inventory, ignition sources, passive fire protection measures, active fire protection systems, items outside the fire compartment, and field verification.

5.4. Supervision of Construction

After the issuance of the construction permit, the regulatory body supervises the construction of the facility. In order to get a sufficiently detailed picture of he implementation of the fire protection arrangements at the nuclear facility, the applicant shall furnish the regulatory body with accounts of the following items:

(1) Wall, floor and ceiling structures of the fire boundaries.
(2) Fire doors and hatches and their fire resistance.
(3) Types and fire resistance of fire stops used in cable and pipe penetrations.
(4) Fire detection and alarm system.
(5) Fire extinguishing systems.
(6) Fire venting and smoke extraction.
(7) Removal of extinguishing water.
(8) Emergency lighting.

The results of the fire hazards analyses shall be reported comprehensively to facilitate the assessment of the fire resistance of the structures.

5.5. Operating Permit

For this stage, the Final Safety Analysis Report has to be submitted to the regulatory body. In addition, accounts of items relating to the planned fire protection arrangements shall also be submitted.

All the organizations which might be called to a fire, should be consulted at the planning stage to avoid the need for later adoption of protective measures at increased cost and delay.

A fire prevention and protection organization should be established as an integrated department of plant management and provided sufficient responsibility, authority and manpower to permit effective performance on a 24 hour day basis.

The areas of consideration of facility management in designing their preplanning for an emergency program should include as a minimum the following requirements.

(1) A self inspection program.
(2) An emergency and fire fighting organization with an outline its training program.
(3) Personnel control as it relates to emergency situations.
(4) Health Physics group responsibilities.

(5) A coordinated response plan with public emergency forces (Civil Defense .., Ministry of Health ......) including periodic drills.
(6) Procedures for loss minimization and decontamination.
(7) The safeguarding of valuable process data and records.
(8) Community relations.

The self inspection program should be formal and conducted objectively by knowledgeable employees who have a good understanding of the hazards to be safeguarded. The reports of these inspections should be reviewed by management at a level which can initiate corrective action. The self inspection report forms should be specifically designed for each facility and include all aspects of basic fire protection as well as those unique to the facility.

Fire emergency procedures shall be established for all personnel. Sufficient training shall be conducted to ensure that every person is familiar with the emergency procedures and his assigned responsibilities.

Drills shall be held at least quarterly, but should be held more frequently as operations permit, especially in the case of buildings involving high fire risk.

5.6. Fire Program

The fire protection manager is responsible for the implementation of the fire program. This program usually encompasses both prevention and protection and should include the following:

(1) Interpretation of applicable codes, regulations and standards.
(2) Design review of the initial fire protection system, alterations and extensions.
(3) Review of and consultation on fire safety aspects of process changes.
(4) Issue of hot work permits (welding, cutting, .. etc.).
(5) Inspection of equipment for fire hazard.
(6) Relationships with insurers (where applicable).
(7) Liaison with official safety organizations.
(8) Fire protection equipment inspection and maintenance.
(9) Fire brigade organization and training.
(10) Emergency fire procedures.
(11) Fire and damage investigations and reports.
(12) Supervision during impairment of protection (and notification to insurers if applied).
(13) Emergency planning for minimization of effects of damage.

5.7. Supervision During Operation

The regulatory body supervises the inservice inspections performed by the owner of the facility to the extent deemed necessary and carries out inspections relating to fire protection in accordance with its own program.

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UPGRADING PROGRAMMES

(Session 6)

Chairperson

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UPGRADING OF FIRE PROTECTION ARRANGEMENTS AT MAGNOX POWER STATIONS IN THE UNITED KINGDOM

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Abstract

The methodology used in conducting fire hazard assessments at Magnox Reactor power stations operated by Magnox Electric plc is described. The assessments use a deterministic approach. This includes the identification of essential plant and the associated supporting systems required for the safe trip, shutdown and post-trip cooling of the reactor, assessment of the location of the essential plant and the vulnerability of these plant in the presence of a fire, assessment of essential functions against the effects of a fire and identification of improvements to the fire protection arrangements. Practical aspects of fire protection engineering on operating power stations are discussed and examples of improvements in protection described.

1. INTRODUCTION

The Magnox Reactors were designed and constructed between 1950s and early 1970s. They are fuelled with natural uranium in a graphite-moderated core and cooled by circulation of pressurised carbon dioxide gas. There are a total of nine Magnox Reactor power stations in the U.K. six of which are in operation and three being decommissioned. Four of the six operating stations use steel pressure vessels and two use pre-stressed concrete pressure vessels.

During its assessment of Long Term Safety Reviews (LTSRs), HM Nuclear Installations Inspectorate (NII) identified fire hazard as one of the twelve Generic Issues which they wished to be addressed for all the Magnox stations, with a view to obtaining improvements to safety:

"Consider whether any improvements to fire zoning and equipment are available and confirm the extent to which the installed system complies with modern standards".

In order to ensure the adoption of a common and systematic approach to identification of fire hazards and improvements to fire protection arrangements, a company strategy on fire safety issues was developed which formed the basis for fire hazard assessments of all the Magnox power stations.

2. MODERN STANDARDS

The principles followed in the design of a modern nuclear power station are given in [1]. The purpose of these principles is to ensure that adequate reactor protection, instrumentation and essential systems will always be available in the event of a fault such as a fire. In the 1980s, the CEGB, Magnox Electric's predecessor, produced specific safety guidelines for the design of its newer power stations [2, 3]. These included the need to provide fire detection and suppression systems of appropriate capacity and to avoid or reduce combustible materials at its nuclear power stations wherever possible. Benefits were also claimed from the use of physical fire barriers which divided a power station into several fire zones. Fire protection provided in the latest nuclear power stations in the U.K. were based on extensive segregation of essential plant, limitation of the amount of combustible material contained within each segregated area, backed-up by detection and suppression systems. The guidelines on fire protection in nuclear power plants were also issued by the IAEA [4].

In summary, the modern standard for fire protection is based on the elimination or reduction of combustible materials, the establishment of fire zones using appropriate fire barriers to segregate the redundant essential plant, the provision of adequate fire detection and suppression systems and implementation of a sound management system to ensure that a high standard of fire safety is maintained.

3. FIRE PROTECTION PHILOSOPHY

The safety principle for the review of fire protection at Magnox power stations is that:

> "for any fire in a given location, sufficient plant should be available to ensure safe trip, shutdown and post-trip cooling of the reactor".

The philosophy of the fire protection to maintain this safety principle is that of defence in depth. This includes the following objectives:

1) Prevention of fires starting. Plant operation and all modification work should be carried out to minimise the probability of fires starting. This should include minimisation of combustible materials in the vicinity of plant and good housekeeping.

2) Early detection and suppression of fires. Early detection would minimise the use of suppression systems and therefore lead to reduced plant damage. To achieve this, adequate and appropriate provision of fire detection and suppression systems is required.

3) Prevention of fires spreading. This is achieved by using appropriate passive fire protection such as fire barriers and dampers. The spread of a fire can also be minimised if fire resistant materials are used wherever possible.

4) Provision of effective fire fighting teams. This is essential if automatic fire suppression systems fail to extinguish the fire.

It is important that confidence can be demonstrated in the ability of fire detection and suppression systems to meet the demands likely to be made upon them.

During the LTSRs, this was done by assessing the vulnerability of the systems to single random failures which may be overcome by installing redundant or diverse detectors.

Diverse power supplies were provided for the detection systems such as dedicated batteries or the station-guaranteed electrical supplies. The new detection systems are normally analogue addressable systems capable of functioning even with a break in the detection loop from the control box. These precautions provide assurance that fires will be detected.
The appropriate British Standards were used as the basis of the modern standards. They covered both active and passive protection systems. The differences between the original and current standards were assessed against nuclear safety. Where shortfalls were found, improvements were undertaken. The aim of the improvements were to achieve fitness for purpose.

4. FIRE SAFETY ASSESSMENT STRATEGY AND CRITERIA

The fire safety assessments were focused on plant and equipment that was essential to achieve nuclear safety. They were not concerned with means of escape or personnel safety in the event of fire, which was covered by the station fire certificates. The assessments also made no reference to the commercial risk of fire.

The following strategy is used in the fire safety assessment.

1) Identification of plant required for the safe trip, shutdown and post-trip cooling of the reactor. This would include the plant that was essential for nuclear safety, their support services (e.g. essential electrical supplies) and the plant that was desirable, in the sense that its availability would assist in maintaining nuclear safety.

2) Identification of the locations of essential plant and support services. This would identify fire barriers and define fire zones to be assessed.

3) Assessment of the adequacy of fire protection of fire zones, containing essential plant and supporting services against assessment criteria (see below).

4) Identification of improvements required to satisfy the safety principle given in section 3.

The criteria for fire protection assessment of mutually redundant plant are:

a) Segregated by fire barriers of 3 hr rating, <u>or</u>

b) Separated by 6 metres horizontally with no intervening combustibles and the fire zone should have an automatic detection and suppression system, <u>or</u>

c) Separated by fire barriers of 1 hr rating and the fire zone should have an automatic detection and suppression system, <u>or</u>

d) Protected by two diverse methods of fire detection and a fast acting fire suppression system, <u>or</u>

If the above are not met, a fire hazard analysis is required to justify the existing arrangements against safety principle and, if necessary, propose improvements to either fire protection or the plant.

The process involved a systematic approach with each plant area and essential function assessed in turn. This allowed decision to be made without an extremely detailed study of fire growth.

5. FIRE ASSESSMENT DURING THE LTSR

A detailed assessment of fire hazards was carried out for each Magnox station during the LTSR. The assessment was based on the strategy outlined above.

5.1 Cable Races and Risers

Although the Magnox stations were designed with some segregation of essential plant, some areas for improvement were identified. A common problem was the lack of segregation between essential electrical cabling. This implied that a fire in a cable tunnel would disable all the essential plant (it should be noted that even in such an event a robust safety case can be made for all stations based on natural circulation of the core and forced cooling of the boilers by the tertiary feed system). It was judged that the separation and re-routing of cables through dedicated fire resistant tunnels would entail major plant modification and was not practicable. As a result, greater emphasis in the fire hazard safety case was placed on the detection and suppression systems. Based on criterion 4 (see section 4), fast acting sprinkler protection was provided in the essential cable routes. Linear Heat Detection Cable (LHDC) systems were used to provide early fire detection and were interfaced with pyrotechnic devices which would be activated to break the frangible bulbs on sprinkler heads to initiate water spray. The frangible bulbs in the sprinkler system would operate in the normal manner and act as diverse fire detection. At some stations, smoke detectors were also installed in cable routes as another diverse line of fire detection and were interfaced with fire alarms only.

5.2 Essential Diesel Generators

At some Magnox stations, all the essential diesel generators were housed within the same building and segregation between these diesel engines was not provided. Fire initiated from one diesel generator would be likely to spread to the other generators due to the storage of diesel fuel in such buildings. Improvements were made by:

i) Installing fire resistant fire barriers between diesel engines.
ii) Relocating some diesel engines to a separated area of the site.

For example, the diesel generator house at one power station originally contained five generator sets without segregation. Following the LTSR, diesel sets 2 and 4 were moved to a different location and segregated by a fire barrier. Within the original diesel house, a Durasteel fire barrier was erected between diesel sets 1 and 3 and another between sets 3 and 5.

5.3 Fire Detection

In order to achieve modern standards, extensive survey and assessments were carried out for each Magnox station to identify the need to extend fire detection in the areas where essential plant were located. This was to comply with the criteria outlined in Section 4.

In most cases this was achieved by installing additional smoke detectors to the existing systems. Where fire protection systems did exist, new systems were installed. Additional protection was also provided to suit the need of a particular area. This included heat detectors, flame detectors, etc.

5.4 Passive Protection

During the survey of the Magnox stations, defects in passive fire protection were identified. A significant amount of work was carried out to improve passive protection. This included installation of fire barriers in the areas where such improvement would result in significant safety benefits, for example, segregation of particular cables within the cable routes. Where complete segregation was not practical, fire screens were installed to protect equipment from radiant heat of a fire of its neighbour.

Fire doors of at least three hour rating were installed to protect redundant plant. All the penetrations were sealed using sealant of the same fire resistance as the walls themselves.

5.5 Control of Improvements

Improvements in fire protection were implemented in accordance with Magnox Electric's plant modification procedures. Proposals for modifications were presented in written submissions which took into account both the nuclear safety implications of the modification and consequences of improper design and installation. Submissions were categorised based upon their nuclear safety significance and were subject to internal independent assessment. The submissions could be called for further assessment by the British Nuclear Regulator NII. Practical work on site was covered by formal quality assurance programmes described in details in Method Statements and controlled by Permits for Work issued by the stations. These measures were designed to ensure that the work could be carried out safely without endangering either plant equipment or the personnel involved.

6. PROBLEMS ENCOUNTERED AFTER IMPROVEMENTS

Teething problems are often inevitable for major modification projects. Improvements to fire protection were no exception. This was largely due to the scale and complexity of the work. Most problems encountered after the improvements were minor and were resolved with relative ease. There were occasions when more investigations were required in order to solve the problems.

One of the problems encountered following the backfitting work was spurious operation of the LHDC system. At one station, such spurious operation led to activation of sprinklers.

During the investigation of the problems, it was discovered that the LHDC system picked up noise signals from external sources. The likely routes for picking up such noise signals were the cabling and LHDC local units. The following modifications to the LHDC system were carried out:

i) The existing multi-point earthing arrangement was replaced with a single-point earth to minimise the noise signals generated in the cabling by the difference in electrical potentials at different earth points.
ii) Filters on the inputs to the LHDC local units were improved to minimise the noise signals generated in the local units by the interfacing components.
iii) The existing time delay cards in the LHDC local units were replaced to increase the time delay in order to eliminate the effect of noise signals with durations less than this increased time delay.

Since the modification, the LHDC system has operated satisfactorily and no further spurious operation has been recorded.

7. FUTURE WORK

A large investment has been made by Magnox Electric in improved fire protection hardware since the LTSR. Over ,10 M has been spent during the process of upgrading fire protection at Magnox stations and the work is now largely complete. Commitment to the achievement of fire safety is ongoing as Magnox Electric is committed to comprehensive review of fire safety, as well as other safety issues, on a regular basis throughout the rest of the operating life of its power stations.

During the forthcoming Periodic Safety Reviews of the operating Magnox stations, fire safety will again be reviewed to ensure that all the stations are protected against fire hazards and all fire protection systems are fit for purpose.

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СОСТОЯНИЕ ПОЖАРНОЙ БЕЗОПАСНОСТИ НА АТОМНЫХ СТАНЦИЯХ В РОССИЙСКОЙ ФЕДЕРАЦИИ И ТЕХНИЧЕСКАЯ СТРАТЕГИЯ МИНАТОМА РОССИИ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ

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Abstract-Аннотация

THE FIRE SAFETY OF RUSSIAN NUCLEAR POWER PLANTS AND THE TECHNICAL POLICY OF THE MINISTRY OF THE RUSSIAN FEDERATION FOR ATOMIC ENERGY IN THE FIELD OF FIRE SAFETY.

The author reports that the Russian Federation has nine nuclear power plants in operation, comprising 29 units with a total capacity of 21 242 MW which may be classified as follows according to reactor type: 13 units with pressurized water reactors of type WWER-440 (6 units) and WWER-1000 (7 units); 15 units with uranium-graphite channel-type reactors (11 of which with RBMK-1000 reactors); and one unit with a fast reactor. The organization in charge of operating these plants is the State enterprise, Russian State Concern for the Production of Electrical and Thermal Energy at Nuclear Power Plants (Rosenergoatom). Data are provided on the numbers of fires at nuclear power plants in recent years and the results of follow-up action on fires undertaken by the Ministry of the Russian Federation for Atomic Energy (Minatom) and the operating organization are reported. The basic problems are described as well as the actions taken by the operating organizations to ensure the fire safety of nuclear power plants, taking into account past fires. It is reported that after the fire at Chernobyl nuclear power plant, the Government issued in 1988 the Comprehensive Measures for Increasing the Fire Safety of Nuclear Power Plants. Work on implementing these measures included the treatment of cable runs with fire retardant compounds, the fireproof sealing of cable penetrations through building structures, and the fitting of fire protective collars in cable boxes, while fire hazardous locations were redesigned and the fire resistance limit of containment structures was increased to 1.5 h. Taking 1994 as a datum line, Rosenergoatom elaborated an overall programme for increasing the fire safety of operating nuclear power plants which was approved by the management and agreed with the Board of the State Fire Protection Service of the Russian Ministry of Internal Affairs. The programme takes into account Russian experience in the operation of nuclear power plants and also the recommendations contained in the IAEA Safety Guide No. 50-SG-D2, Fire Protection in Nuclear Power Plants. In nuclear power plant projects with new generation reactors, questions of fire safety are dealt with in accordance with the requirements of the latest fire safety regulations. Taking advantage of experience in increasing the fire safety of operating nuclear power plants in Russia and similar experience in this area in foreign nuclear power plants, Minatom is working closely with the operating organizations to ensure the safe shutdown of reactors in the event of any fire. Finally, the priorities laid down by Minatom for nuclear power plants in the coming years are presented.

СОСТОЯНИЕ ПОЖАРНОЙ БЕЗОПАСНОСТИ АТОМНЫХ СТАНЦИЙ РОССИИ И ТЕХНИЧЕСКАЯ ПОЛИТИКА МИНАТОМА РОССИИ В ОБЛАСТИ ПОЖАРНОЙ БЕЗОПАСНОСТИ.

В докладе сообщается о том, что на территории Российской Федерации действует 9 атомных электростанций. В состав этих АЭС входит 29 энергоблоков суммарной мощностью 21 242 МВт, которые по типам реакторных установок делятся на следующие группы: 13 энергоблоков с корпусными водо-водяными реакторами типа ВВЭР-440 (6 блоков) и ВВЭР-1000 (7 блоков), 15 энергоблоков с уран-графитовыми канальными реакторами (из них 11 с реакторами РБМК-1000), 1 энергоблок с реактором на быстрых нейтронах. Функции эксплуатирующей организации возложены на государственное предприятие "Российский государственный концерн по производству электрической и тепловой энергии на атомных станциях" (концерн "Росэнергоатом"). Приводятся данные о количествах пожаров на атомных электростанциях в последние годы и сообщается о результатах действий Министерства по атомной энергии Российской Федерации (Минатом) и эксплуатирующей организации по фактам случившихся пожаров. Раскрываются основные задачи и действия эксплуатирующих организаций по обеспечению пожарной безопасности атомных электростанций с учетом происшедших пожаров. Сообщается о том, что после пожара на Чернобыльской АЭС по поручению правительства в 1988 году был разработан и утвержден документ "Сводные мероприятия по повышению пожарной безопасности атомных станций". В ходе выполнения этих мероприятий проведена обработка кабельных трасс огнезащитными составами, выполнены огнезащитные уплотнения кабельных проходок через строительные конструкции, огнезащитные пояса в кабельных коробах, проведена реконструкция пожароопасных помещений с доведением предела огнестойкости ограждающих конструкций до 1,5 часа и пр. С учетом сложившихся обстоятельств в 1994 году концерном "Росэнергоатом" была разработана отраслевая перспективная комплексная программа повышения пожарной безопасности действующих АЭС, которая утверждена руководством концерна и согласована с Главным управлением Государственной противопожарной службы Министерства внутренних дел Российской Федерации. Программа учитывает российский опыт эксплуатации атомных станций, а также рекомендации Руководства МАГАТЭ по безопасности № 50-SG-D2 "Противопожарная защита на атомных электростанциях". В проектах АЭС с реакторами нового поколения вопросы пожарной безопасности решаются в соответствии с требованиями современных документов по пожарной безопасности. Минатом России с учетом опыта работы по повышению пожарной безопасности действующих АЭС и опыта организации этой работы на зарубежных атомных станциях направляет работу эксплуатирующих организаций на обеспечение безопасной остановки реактора при любом пожаре. В докладе указаны приоритеты, установленные для АЭС Минатомом России на ближайшие годы.

В системе Министерства Российской Федерации по атомной энергии имеется большое количество предприятий: имеющих радиационную и ядерную опасность. К ним относятся предприятия с ядерными установками промышленного или исследовательского назначения и, в том числе атомные электростанции. Всего в составе Российской Федерации после распада Советского Союза осталось девять атомных электростанций (Балаковская АЭС, Белоярская АЭС, Билибинская АЭС, Калининская АЭС, Кольская АЭС, Курская АЭС, Нововоронежская АЭС, Смоленская АЭС, Ленинградская АЭС).

Атомная энергетика России дает более 12% всей вырабатываемой в стране электроэнергии.

В 1992 году по Указу Президента Российской Федерации от 07.09.92 г. № 1055 функции эксплуатирующей организации атомных электростанций возложены на государственное предприятие "Российский государственный концерн по производству злектрической и тепловой энергии на атомных станциях" (концерн "Росэнергоатом"), входящий в состав Министерства Российской Федерации по атомной энергии. В состав концерна вошли все вышеперечисленные атомные электростанции, кроме Ленинградской АЭС.

Деятельность концерна направлена на обеспечение безопасного и эффективного производства энергии на АЭС, обеспечение ядерной , радиационной и пожарной безопасности.

В Министерстве Российской Федерации по атомной энергии имеется Департамент безопасности, экологии и чрезвычайных ситуаций, который ответственен за организацию работы по обеспечению пожарной безопасности на всех предприятиях Министерства, включая и атомные электростанции. По роду деятельности Департамент участвует во всех организационных мероприятиях по обеспечению пожарной безопасности на всех предприятиях .

Для обеспечения противопожарной защиты на всех ядерноопасных и других предприятиях Министерства имеются подразделения Государственной противопожарной службы МВД России с численным составом более 5000 человек, осуществляющих как работу по профилактике пожаров на предприятиях отрасли и на АЭС, так и их тушение.

Министерство, в тесном взаимодействии с Государственной противопожарной службой и концерном "Росэнергоатом" в целях обеспечения противопожарной защиты предприятий организует и проводит работу по следующим направлениям;

- организация и обеспечение административного контроля за противопожарным состоянием объектов Министерства,
- участие в разработке отраслевых нормативных документов по пожарной безопасности,
- организация пожарно-профилактической работы,
- проведение анализов противопожарного состояния производств,
- разработка и применение надежных и эффективных систем обнаружения и тушения пожаров,
- Применение для защиты кабельных потоков на предприятиях и АЭС отечественных и зарубежных огнезащитных материалов и другое.

Указанная работа проводится в соответствии с требованиями российских, межотраслевых и отраслевых нормативных документов;

- федеральным законом "Об использовании атомной энергии",
- федеральным законом "О пожарной безопасности",
- ГОСТом 12.1.004-91 "Пожарная безопасность. Общие требования",
- правилами пожарной безопасности в Российской федерации (ППБ-01-93),
- общими положениями обеспечения безопасности атомных станций (ОПБ-88),
- основными правилами обеспечения эксплуатации атомных станций (ОПЭ АС),
- противопожарными нормами проектирования атомных станций ВСН 01-87, правилами пожарной безопасности при эксплуатации атомных станций (ППБ АС-95).

ТАБЛИЦА 1. КОЛИЧЕСТВО ПОЖАРОВ НА АЭС РОССИИ В 1994 - 1996 гг.

АЭС	Количество пожаров		
	1994 г.	1995 г.	1996 г.
Балаковская	-	-	-
Белоярская	2	-	-
Билибинская	-	-	-
Калининская	-	-	2
Кольская	2	-	-
Курская	2	4	-
Нововоронежская	2	5	-
Смоленская	1	2	-
Ленинградская	-	1	-
ВСЕГО	9	12	3

Министерством в 1996 году проведено заседание отраслевой комиссии по чрезвычайным ситуациям на тему “О состоянии работы по замене сгораемого утеплителя кровель машинных залов АЭС на несгораемый”. Замена сгораемого утеплителя на несгораемый предложена «Сводными мероприятиями по повышению пожарной безопасности АЭС» (СМПБ-88).

Для проведения постоянной работы по обеспечению пожарной безопасности на атомных станциях созданы Центральные пожарно-технические комиссии, в состав которых включены руководители основных подразделений, специалисты по пожарной безопасности, водоснабжению, противопожарной автоматике, кабельному хозяйству, вентиляции и противодымной защите и специалисты пожарной охраны. Работа комиссии проводится в соответствии у утвержденными руководством АЭС планами.

На всех атомных станциях созданы подразделения по пожарной безопасности из числа сотрудников АЭС, численностью от 1 до 3 человек. Как правило это специалисты, имеющие большой опыт практической работы по пожарной безопасности.

Для поддержания в рабочем состоянии систем пожарной сигнализации и установок автоматического пожаротушения в составе электроцехов созданы участки по техническому обслуживанию и ремонту указанных систем.

На всех атомных электростанциях ежегодно проводится анализ работы за прошедший год и намечаются мероприятия по повышению пожарной безопасности.

ТАБЛИЦА 2. КОЛИЧЕСТВО ПОЖАРОВ И ПОЖАРООПАСНЫХ НАРУШЕНИЙ В РАБОТЕ ОБОРУДОВАНИЯ АЭС ПО ОТДЕЛЬНЫМ АЭС РОССИИ И ТИПАМ РУ В 1994 - 1996 гг.

АЭС, блок, РУ	1994 г.	1995 г.	1996 г.
Бал АЭС-2, ВВЭР-1000	2	-	1
Бал АЭС-4, ВВЭР-1000	3	-	-
Клн АЭС-1, ВВЭР-1000	2	1	1
Клн АЭС-2, ВВЭР-1000	-	-	3
Ко АЭС-1, ВВЭР-440	1	-	-
Ко АЭС-3, ВВЭР-440	1	-	-
Ко АЭС-4, ВВЭР-440	1	-	2
Ку АЭС-2, РБМК-1000	3	1	2
Ку АЭС-3, РБМК-1000	-	2	-
Ку АЭС-4, РБМК-1000	2	1	-
Лн АЭС-1, РБМК-1000	-	-	1
Лн АЭС-2, РБМК-1000	-	1	-
Лн АЭС-3, РБМК-1000	1	-	-
Лн АЭС-4, РБМК-1000	-	-	1
НВ АЭС-3, ВВЭР-440	1	-	1
НВ АЭС-5, ВВЭР-1000	2	3	1
См АЭС-2, РБМК-1000	-	1	-
См АЭС-3, РБМК-1000	1	1	-
Бел АЭС-3, БН-600	2	-	-

ТАБЛИЦА 3. ВЫПОЛНЕНИЕ СВОДНЫХ МЕРОПРИЯТИЙ ПО ПОЖАРНОЙ БЕЗОПАСНОСТИ СМПБ-88

АЭС	Количество мероприятий			% выполнения
	Подлежало выполнению	Выполнено	Не выполнено	
Балаковская	18	16	2	88,8
Белоярская	20	15	5	75,0
Билибинская	19	10	9	52,6
Калининская	19	17	2	89,5
Кольская	24	19	5	79,2
Курская	29	20	9	69,0
Нововоронежская	28	20	8	71,4
Смоленская	22	15	7	68,2
ВСЕГО	179	132	47	73,7

Однако несмотря на проводимую работу по обеспечению пожарной безопасности на АЭС пожары имеют место. За 1994-1996 годы на атомных станциях России произошло 24 пожара и возгорания. Подавляющее большинство пожаров произошло из-за следующих причин:

- нарушение требования правил пожарной безопасности (в основном несоблюдения правил производства огневых и пожароопасных работ),
- неисправность технологического оборудования,
- нарушение режима курения,
- короткие замыкания.

Среди перечисленных причин пожаров на неисправность технологического оборудования приходится около 36% пожаров, а на короткие замыкания около 26%.

По годам пожары распределились следующим образом(см. таблицу №1):

В 1997 году пожаров на АЭС не зарегистрировано.

Приведу два примера пожаров, происшедших по технологическим причинам:

17 февраля 1996 года в 09 час. 39 мин. на энергоблоке № 2 Калининской АЭС произошло возгорание промасленной теплоизоляции на участке между блоком регулирования и паровыпуском турбонасоса 2 ТПН-1 на площади около 0,1 кв.м.

Пожар ликвидирован персоналом турбинного цеха до прибытия дежурного караула пожарной части.

Причина пожара заключалась в недостаточной эффективности системы отсоса масляных паров со сливного маслопровода 2 ТПН-1 (вследствии наличия контруклона на сливном маслопроводе и трубопроводе отсоса на эксгаустеры). Произошла их утечка через уплотнение вала и накопление на теплоизоляции паровыпуска турбопривода ОК-12А 2 ТПН-1. Воспламенение масла на теплоизоляции произошло вследствие прямого контакта теплоизоляции с горячей поверхностью ОК-12А.

04 апреля 1996 года в 06 час. 05 мин. на блоке № 2 Курской АЭС произошло возгорание масла, попавшего на изоляцию под ЦВД в результате течи по сварному шву трубы подвода силового масла в полость сервомотора стопорного клапана №1 (СК-1) турбогенератора № 3. Пожар локализован персоналом турбинного цеха и ликвидирован дежурным караулом пожарной части.

Причиной пожара явилось механическое повреждение- появление трещины в зоне сварного шва соединения выключателя сервомотора СК-1 с трубой подвода силового масла.

Каждый случай пожара на АЭС тщательно расследуется. Выясняются причины его возникновения и рассматриваются варианты и возможные последствия для безопасности АЭС в целом. Разрабатываются мероприятия для предупреждения возникновения подобных инцидентов в будущем.

Для полноты картины и более подробного анализа противопожарного состояния атомной станции целесообразно рассматривать не только пожары , но и пожароопасные нарушения в работе оборудования , которые потенциально могли привести к пожарам. Различные нарушения представляют собой разную степень опасности для АЭС и поэтому требуют тщательного и всестороннего анализа с точки зрения всех вариантов возможных последствий каждого из нарушений. Риск наступления нежелательных последствий от каждого нарушения увеличивается с ростом частоты последних.

Количество пожаров и пожароопасных нарушений на основном оборудовании АЭС за 1994-1996 годы представлено в таблице № 2.

Выполнение Сводных мероприятий по пожарной безопасности АЭС

Авария в 1986 году на Чернобыльской АЭС и проведенные после нее комиссионные проверки всех атомных электростанций СССР показали многие недостатки в их противопожарной защите. Поэтому по поручению Совета Министров СССР от 09.07.87 года № Щ-2053 и в целях совершенствования противопожарной защиты АЭС были разработаны “Сводные мероприятия по повышению пожарной безопасности атомных станций” (СМПБ-88), введенные в действие приказом по Министерству по атомной энергии СССР от 16.02.88 № 6 дсп. СМПБ-88 предусматривали выполнение ряда мероприятий , способствующих значительному повышению пожарной безопасности АЭС. Они включали в себя:

- замену сгораемых кровель машинных залов на несгораемые,
- замену устаревшего оборудования и систем пожарной автоматики,
- поиск и разработку новых материалов с повышенными огнезащитными и огнестойкими свойствами,
- строительство тренировочных полигонов для пожарной охраны и пожарных депо для пожарной техники,
- обработку кабельных потоков огнезащитными материалами и др.

Для реализации указанных мероприятий Правительством были даны конкретные поручения другим министерствам и ведомствам на выполнение научно-технических работ по разработке нового пожарно-технического оборудования, систем раннего обнаружения и тушения пожаров, а также по развитию производства материалов с повышенными огнезащитными свойствами, огнезащитных покрытий для повышения огнестойкости несущих металлоконструкций.

В ходе выполнения этих мероприятий за последние годы на АЭС проведена обработка кабельных трасс огнезащитными составами, выполнены огнезащитные уплотнения кабельных проходок и огнезащитные пояса в кабельных коробах , проведена реконструкция дверей и перегородок кабельных и других пожароопасных помещений с доведением их предела огнестойкости до 1,5 часа, выполнено разделение кабельных тоннеле на отсеки протяженностью не более 72 метров для реакторов типа РБМК и 50 м. для ВВЭР. Проведена реконструкция пожарного водопровода с выделением трубопровода высокого давления и заменой чугунной арматуры на стальную, на водохранилищах оборудованы пирсы для забора воды пожарной техникой. Построены новые здания пожарных депо на Кольской, Калининской и Курской АЭС и боксы для размещения крупногабаритной пожарной техники на Белоярской, Балаковской, Калининской, Курской и Нововоронежской АЭС, а также пожарные полигоны для тренировок личного состава пожарных частей и персонала на Балаковской , Белоярской и Курской АЭС.

Для огнезащитной обработки кабельных потоков на АЭС применяются как огнезащитные покрытия, разработанные институтом НИКИМТ и Сибирским филиалом ВНИИНМ Минатома России, так и иностранных фирм, как например, СФТ Брандшутц (Германия) (Балаковская АЭС, Ленинградская АЭС).

Выполнение работ по замене сгораемых кровель машзалов на несгораемые на АЭС :

На Калининской АЭС выполнены работы по замене сгораемых кровельных панелей на несгораемые на 1 и 2 блоках.

На Нововоронежской АЭС разработан проект на замену сгораемой кровли на 5 энергоблоке. Работы начаты в этом году.

На Курской АЭС работы выполнены на площади 1505 кв.м. на энергоблоке № 1.

Для Кольской АЭС проведены экспериментальные работы по замене кровельных панелей машзала на площади 222 кв.м. и вентцентра на площади 36 кв.м.

В качестве компенсирующих мероприятий на кровлях машзалов со сгораемым утеплителем выполнены противопожарные разрывы шириной от 6 до 24 метров из несгораемых материалов, проложены сухотрубы с кольцевой разводкой по кровле, смонтированы дополнительные пожарные лестницы.

Для обеспечения выполнения работ по замене сгораемых кровель машзалов концерн "Росэнергоатом" начато строительство предприятия по производству негорючего утеплителя "Диатем". В настоящее время идут пуско-наладочные работы.

Налажено производство трудносгораемого утеплителя "Изолен", который разрешен Главным управлением противопожарной службы МВД России к применению для кровель машзалов АЭС.

На Кольской АЭС теплогенератор № 6 переведен на пожаробезопасное масло ОМТИ.

На Балаковской АЭС, в условиях действующих энергоблоков проведены работы по устройству систем дымоудаления из помещений, не имеющих ограничений по связи с окружающей средой. Работы проводились с использованием взрывных технологий, которые выполнялись Всероссийским научно-исследовательским институтом экспериментальной физики (ВНИИЭФ).

Произведена замена горючего пластиката 57-40 на путях эвакуации персонала в зоне строгого режима:

На Балаковской АЭС - блок № 1, 2 (Макро-АС)

- блок № 3 (ЭП-5264)
- блок № 4 (ЭК-01)

На Смоленской АЭС - на всех блоках (ЭК-01, шлифованный бетон).

На Кольской АЭС - в лестничных клетках на площади 2562 кв.м. на основе ЭП-5264.

На Нововоронежской АЭС - на площади 146 кв.м. на 4 энергоблоке проводится опытная эксплуатация наливных полов из материала Макро-АСТ.

На Калининской АЭС - работы выполнены на площади 800 кв.м. из материала фирмы «Перматекс».

На Курской АЭС - работы выполнены на площади 450 кв.м. из материала Е-101 «Германия».

Как компенсирующее мероприятие в покрытиях полов выполнены противопожарные разрывы из нержавеющей стали (Кольская АЭС, Курская АЭС), метлахской плитки (Билибинская АЭС), шлакоситалловых плит (Белоярская АЭС).

Но многие мероприятия программы обеспечения пожарной безопасности АЭС не выполнены или находятся в стадии выполнения. Конкретно о невыполненных требованиях можно сказать следующее:

1. «Выполнить системы резервного энергоснабжения с резервным щитом управления и контроля параметров реакторной установки»

На Курской АЭС в настоящее время завершены основные строительные работы на КНП-1, ведутся отделочные работы, производится монтах вентсистем и вентагрегатов, аккумуляторных батарей. На 90% смонтирована система пожаротушения.

Срок окончания работ -1997 год (1 эн.блок) -1999 год (1 эн.блок)

На Нововоронежской АЭС ведутся строительно-монтажные работы помещений под установку аккумуляторных батарей на ДГС 3-4 блоков, выполнена комплектация металлоконструкций, размещен заказ на изготовление железобетонных конструкций . Поставка аккумуляторных батарей предполагается в августе 1998 года и будет осуществляться по счету «Ядерная безопасность», финансируемому «Европейским банком реконструкции и развития» (ЕБРР). Разработана проектная документация и размещен заказ на изготовление оборудования резервного щита управления и контроля параметров РУ энергоблоков № 3 и 4.

Срок окончания работ -1999 год.

2. «Выполнить реконструкцию установок пожарной сигнализации с заменой приемных станций и извещателей морально устаревших на современные»

На Курской АЭС данная работа включена в единый рабочий проект и в настоящее время проводится комплектование оборудования и монтах кабельных конструкций для автоматической пожарной сигнализации. Приемные станции и пожарные извещатели укомплектованы полностью.

Срок завершения работ -1997 год (1 эн.блок) -1999 год (2 эн.блок).

На Смоленской АЭС техническое задание на проектирование в стадии разработки и согласования. Ведутся работы по поставке оборудования швейцарской фирмой «Церберус». Планируемый срок выполнения мероприятия 1997-2000 годы (1 эн.блок) , 1999-2001 годы (2 эн.блок).

3. «Произвести замену кровельных панелей со сгораемым утеплителем в покрытиях машинных залов на панели с несгораемым или трудносгораемым утеплителем»

На Балаковской АЭС , Кольской АЭС, Курской АЭС оформлены договора на поставку несгораемого утеплителя «Диатем» и планируется поэтапная замена сгораемого утеплителя в кровлях машзалов. Ориентировочно сроки окончания работ 1998- 1999 годы.

На Курской АЭС горючий утеплитель ПСБ-С заменен на трудносгораемый утеплитель «Фенопласт» на участке кровли машзала 1 энблока на площади 4500 кв.м. Заключен договор с АО «Энергоатомпромстрой» на поставку негорючего утеплителя «Диатем». Срок выполнения работ -1997 год (1 эн.блок) -1999 год (2 эн.блок).

На Нововоронежской АЭС разработан проект реконструкции кровли эн.блока № 5. Работы по замене сгораемого утеплителя планируется провести в 1999 году.

На Смоленской АЭС выполнен опытный участок кровли с негорючим утеплителем. Планируемый срок выполнения мероприятия: 1999 год -1 эн.блок , 2000 год - 2 эн.блок.

4. «Оборудовать установками автоматической пожарной сигнализации помещения щитов управления и автоматизированного управления технологическим процессом (УВС,УКТС,АКНП и др.), а также релейных щитов»

На Курской АЭС приемные станции и пожарные извещатели укомплектованы полностью. Выполняются работы по монтажу пожарных извещателей в помещении БЩУ-1. Срок выполнения работ 1997 год - 1эн.блок , 1999 год - 2 эн.блок.

5. «Выполнить системы подпора воздуха в помещениях щитов управления и автоматизированных систем управления технологическим процессом (УВС,УКТС,АКНП,СУЗ и др.), а также в лестничных клетках зданий высотой более 30 м.»

На Курской АЭС в связи с конструктивными особенностями компоновки помещений БЩУ,ЦЩУ,СКАЛА мероприятия выполнить невозможно (заключение Ленинградского отделения АЭП от 14.02.91 г. № 0340/5920-71). В настоящее время системами подпора воздуха оборудованы тамбур-шлюзы щитов управления. Ведутся работы в лестничных клетках и при наличии необходимых средств на оплату оборудования работы будут завершены в установленные сроки.

На Нововоронежской АЭС сданы в эксплуатацию системы подпора воздуха в помещениях щитов управления, лестничных клетках зоны строгого режима и лестничной клетке № 1 эн.блока № 5. Планируемый срок завершения работ -1997 год.

6. «Смонтировать установки водяного пожаротушения главных циркуляционных и питательных насосов, подвалах РДЭС».

На Курской АЭС выполнено на маслохозяйствах ПЭН и ГНЦ с ручным управлением. Перевод в автоматический режим управления будет выполнен при проведении комплекса работ по реконструкции установок пожарной сигнализации. Вмаслохозяйствах РДЭС по согласованию с пожарной охраной выполнена разводка трубопроводов пенного пожаротушения от передвижной пожарной техники.

7. «Завершить перевод установок пожаротушения в автоматический режим работы».

На Белоярской АЭС мероприятие не выполнено из-за большого количества ложных срабатываний пожарных извещателей ДИП-3 (в 1994 году - 269, в 1995 году - 227). В настоящее время рассматривается вопрос применения в установках пожаротушения пожарных извещателей ,разработанных институтом физико-технических проблем Минатома России.

На Курской АЭС ведутся работы по переводу в автоматический режим работы установки автоматического пожаротушения на эн.блоках № 1 и 2 : установлена электрофицированная арматура, смонтированы сборки и шкафы управления задвижками, закончен монтаж кабельных металлоконструкций в машзале и многое другое. Срок выполнения работ 1997-1999 годы.

На Билибинской АЭС работы не выполнены.

8. «Произвести замену горючего пластиката 57-40 на путях эвакуации в зоне строгого режима».

На Белоярской АЭС замена выполнена на площади 1000 кв.м. с применением материала фирмы ПЕРМАТЕКС (Германия). Предполагаемый срок завершения работ - 1999 год.

На Калининской АЭС заключен договор и выполнены работы на площади 1200 кв.м. (общая площадь полов на путях эвакуации в зоне строгого режима -8000 кв.м.). Учитывая финансово-экономическое положение Калининской АЭС, предполагаемый срок выполнения всего объема работ - 2000 год.

На Кольской АЭС выполнена замена пластиката в лестничных клетках аппаратных отделений всех эн.блоков на площади 2562 кв.м. эпоксидным покрытием ЭП-5264. По эн.блокам № 1 и 2 запланирована поставка материалов по счету «Ядерная безопасность», финансируемого ЕБРР. Планируемый срок выполнения работ -1998 год.

По эн.блокам № 3 и 4 работы будут выполняться при наличии финансирования для приобретения материалов.

На Курской АЭС для эн.блока № 1 поставлены в необходимом количестве материалы наливных полов марки 301 фирмы ПЕРМАТЕКС (Германия) и выполнены работы на площади более 2000 кв.м. Сроки окончания работ: 1997 год (1 эн.блок), 1999 год (2 эн.блок), 2000 год (3 эн.блок), 2002 год (4 эн.блок).

На 4 эн.блоке Нововоронежской АЭС выполнен опытный участок наливных полов из материала «Макро-АСТ» на площади 146 кв.м.. Работа по устройству наливных полов на эн.блоках № 3 и 4 включена в счет «Ядерная безопасность», финансируемого ЕБРР, с поставкой материалов по контракту в декабре 1997 года. На эн.блок № 5 разработан проект. Планируемый срок выполнения работ - 1999 год (3 и 4 эн.блоки), - 2000 год (5 эн.блок).

На Смоленской АЭС замена произведена на площади около 3700 кв.м. и ведется постоянно, однако высокая стоимость работ (до 0,5 млн. Рублей за 1 кв.м. площади пола) сдерживает выполнение данной работы на всей необходимой площади. Работы планируется завершить до 2000 года.

8. «Выполнить системы дымоудаления из кабельных и других пожароопасных помещений, эвакуационных коридоров не имеющих ограничений по связи с окружающей средой».

На Билибинской АЭС разрабатывается проектная документация. Выполнение планируется в период реконструкции - в 1999 году.

На Калининской АЭС системы дымоудаления из кабельных помещений 1-й очереди выполнить невозможно по конструктивным особенностям (техническое решение от 09.12.93 г. № 12-5724 подготовлено Нижегородским институтом АЕП, согласовано Калининской АЭС и территориальными органами управления Государственной противопожарной службы.).На системы дымоудаления из эвакуационных коридоров разработан проект и приобретено оборудование, однако монтаж его возможен только в период остановки блоков на ремонт.

Планируемый срок выполнения - 2001 год.

На Кольской АЭС закончены строительные работы по разделению протяженных эвакуационных коридоров противодымными перегородками, ведется поставка оборудования. Для дымоудаления из кабельных и щитовых помещений предполагается использовать передвижные дымососы, поставка которых планируется по программе TACIS-95 в 1998 году. Планируемый срок выполнения мероприятия - 1998 год.

На Курской АЭС ведутся работы на эн.блоке № 1: на 60% смонтированы воздуховоды, устанавливаются вентагрегаты, прокладываются кабельные трассы, ведутся электромонтажные работы. Оборудование укомплектовано на 70 %. Отсутствует часть проектно-сметной документации на системы дымоудаления из машзала, помещений щитов управления и электротехнических помещений. Причиной невыполнения является недопоставка оборудования и переработка проектной документации в связи с принятием решения о размещении в отдельных помещениях маслоохладителей трансформаторов.

Срок завершения работ: 1997 год (1 эн.блок) -1999 год (2 эн.блок) -2003 год (3 эн.блок).

9. «Оборудовать установками газового пожаротушения помещения щитов управления (БЩУ,ГЩУ,ЦЩУ), а также помещения с электронной аппаратурой (УВС,УКТС,АКНП,СУЗ,ВРК,АКРБ и другие) систем автоматизированного управления технологическим процессом АЭС».

Выполнение мероприятия предусматривалось при реконструкции и в зависимости от промышленного производства фреона 13В1, а также работ по разработке малотоксичных установок газового пожаротушения. Этот вопрос не решен в принципе. С одной стороны, в стране отсутствует промышленное производство фреона 13В1, с другой - фреон 13В1 является озоноразрушающим веществом.

Альтернативным решением проблемы предполагалось использование системы аэрозольного тушения пожаров (САТ). Концерном «Росэнергоатом» был заключен договор с институтом «Атомэнергопроект» на разработку такой системы газового пожаротушения и в качестве эксперимента смонтирована установка на Балаковской АЭС.

Однако, разработанные Всероссийским научно-исследовательским институтом противопожарной обороны (ВНИИПО)МВД России и утвержденные Главным управлением Государственной противопожарной службы МВД России нормы НПБ 21-94 «Системы аэрозольного тушения пожаров. Временные нормы и правила проектирования и эксплуатации» допускают применение САТ лишь в качестве дополнительного, но не альтернативного средства автоматической пожарной защиты.

Техническое совещание с участием представителей пожарной охраны и ВНИИПО МВД России приняло решение о подготовке технических требований и рекомендаций для разработки системы газового пожаротушения с учетом специфики АЭС.

До окончания решения вопроса все помещения щитов управления, а также помещений АСУТП оборудованы стационарными углекислотными установками типа 2БР2МА и первичными средствами пожаротушения согласно норм оснащенности.

Процент выполнения противопожарных мероприятий показан в таблице №3.

Техническая политика Минатома России в области пожарной безопасности АЭС

В настоящее время техническая политика Минатома России в направлении обеспечения пожарной безопасности АЭС и других ядерноопасных объектов строится с учетом требований Федерального закона «О промышленной безопасности опасных производственных объектов», принятого Государственной Думой 20 июня 1997 года.

Это требует доведения пожарной безопасности опасного объекта до 100% или до максимально необходимого , что дает право на его эксплуатацию.

Поэтому Минатом России организует работу по обеспечению пожарной безопасности, совместно с Государственной противопожарной службой МВД России таким образом , чтобы добиться выполнения всех противопожарных мероприятий и довести противопожарную защиту объектов до 100% -ной..

Кроме того Минатом России всемерно способствует улучшению технического оснащения подразделений пожарной охраны на особо опасных объектах отрасли путем участия в разработках и исследованиях, проводимых ВНИИПО МВД России, а также в обеспечении производства техники и оборудования для пожарной охраны. В 1997 году Минатом России совместно с другими Министерствами вышел в Правительство Российской Федерации с предложением о налаживании производства для пожарной охраны атомных станций специальных защитных костюмов СОЗ-1.

В целом Министерство поддерживало и будет поддерживать необходимые связи как с Российскими , так и с зарубежными фирмами для обеспечения необходимой и надежной защиты от пожаров особо опасных объектов.

IAEA-SM-345/47

FIRE HAZARD ASSESSMENT OF CANDU PLANTS

XA9847529

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Abstract

The requirements for fire protection of CANDU nuclear power plants have evolved from the rule based requirements applied to the early plants to the performance based standards of the 1990's. The current Canadian standard, CAN/CSA N293 (1995), requires a documented fire hazard assessment to be used in the design of fire detection and extinguishing systems. The Fire Hazard Assessment method uses a standard format for all fire zones in the plant to assess the adequacy of the fire protection measures, first applied to the CANDU 6 design at Wolsong 2/3/4. The grouping of safety related systems into two independent and well separated groups was found to have a large positive impact on the ability to maintain safety functions during a fire. The new CANDU 9 design builds on the experience gained from previous designs, with improvements in grouping and separation and fire protection system design.

1. Introduction

The requirements for fire protection of CANDU nuclear power plants have evolved from the rule-based building code requirements of the 1960's and 1970's to the performance based standards of the 1990's which address the special needs of nuclear power plants. The design and assessment methods for CANDU plants have kept abreast of the rapid development of international requirements for fire protection in nuclear power plants, and have been incorporated in the latest CANDU 9 design.

2. Fire Protection Requirements for CANDU Plants

The requirements for fire protection in the early CANDU plants were primarily to satisfy the National Building Code of Canada, and to take into account the special needs of the nuclear power plant. The building code objectives were to control the combustibility of materials in buildings (based on the type of occupancy), to ensure enough time for people to be evacuated from the building, and to ensure that firefighters or automatic suppression systems were available to extinguish a fire. These objectives were applied through specific rules for fire barriers, ventilation, and fire suppression systems. The nuclear plants were licensed on the basis of these rules, with special exceptions where necessary to accommodate the special safety requirements of a nuclear plant. For example, the requirements for area restrictions and distances to

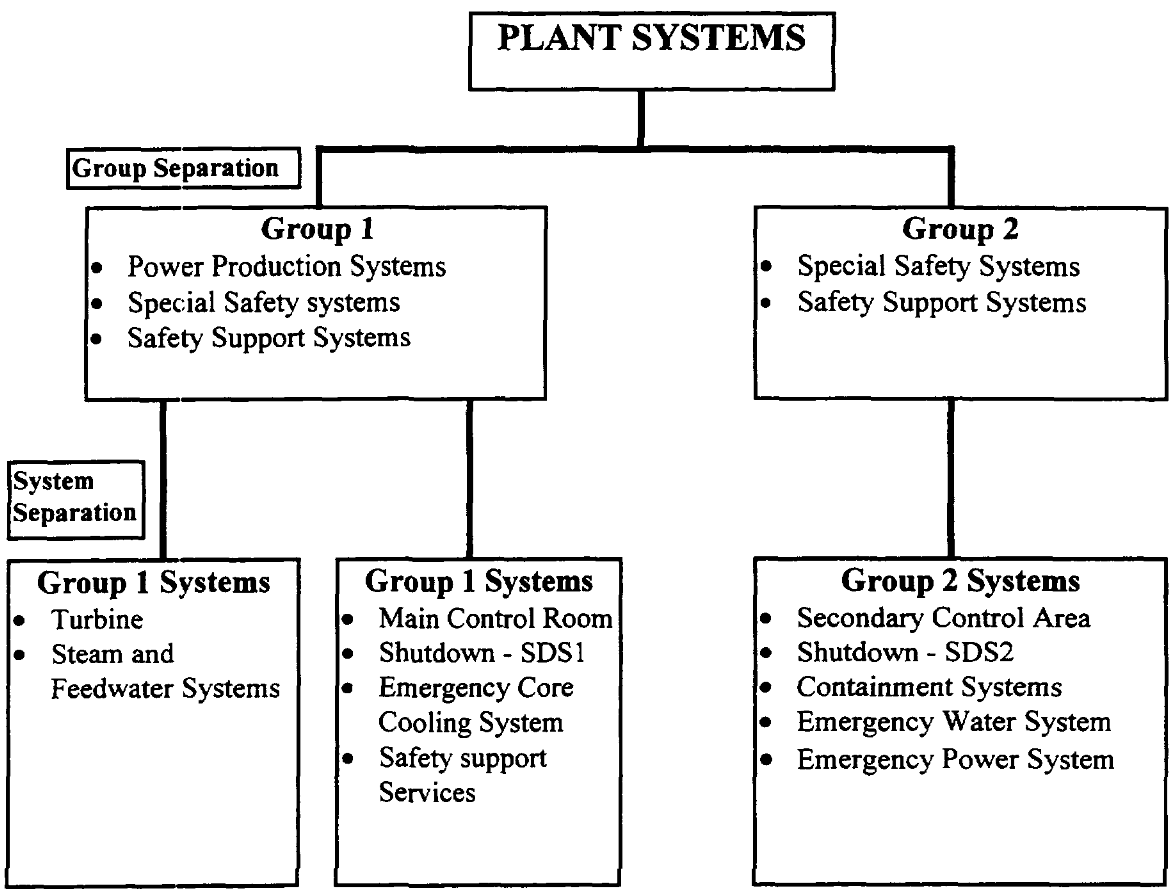

Figure 1. Grouping of Safety Related Systems for CANDU 6 Plants

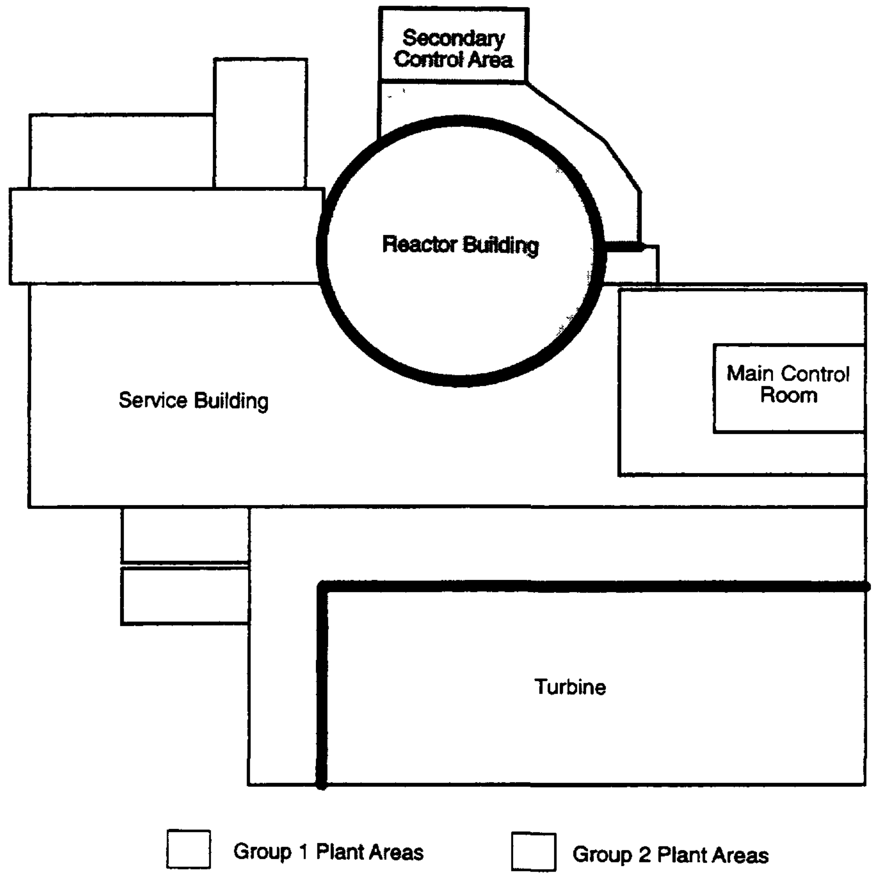

Figure 2. CANDU 6 Layout and major Fire Barriers

exits were changed to reflect the need for a large containment volume and pressure equalization during an accident, and concern about the deleterious effects of automatic sprinkler systems on nuclear instrumentation and power supplies resulted in a greater reliance on manual fire fighting in the early designs. Large open areas were considered acceptable due to the relatively low combustible content in most areas, and the ease of access for fire fighting.

Additional design features ensured that the necessary safety functions could be maintained during and after a fire. These included sufficient water reserves for fuel cooling to permit time for manual mitigating actions by plant staff, and substantial physical separation of cables and systems providing redundant safety functions. The physical separation of systems into two groups is shown in Figure 1 for the CANDU 6 designs Each group has the capability to shut down the reactor, provide fuel cooling, and maintain control and monitoring activities should a fire occur in the other group Figure 2 shows the physical arrangement of those groups in the CANDU 6 plant layout, with the location of the major fire barriers provided in the current design The figure shows the substantial separation of the Group 2 systems outside the reactor building, with very little reliance on the presence of fire barriers to protect against fires occurring in the normally operating Group 1 systems The major fire protection design improvements in the more recent designs have included providing a fire barrier between the two groups of safety related systems outside the reactor building, and upgrading of some fire separations within Group 1, such as the fire barrier between the turbine-generator area and the service building/main control room Other improvements included a much more comprehensive application of automatic fire suppression systems in areas outside of the reactor building, and provision of fire detection and hose stations to all plant areas.

As specific requirements for the fire protection of nuclear power plants evolved in the mid-1980's, the National Standard of Canada, CAN/CSA N293 was developed and issued in 1987. This standard reflected the experience gained in the early CANDU plants, and the principles outlined in IAEA 50-SG-D2 It stressed the minimization of combustibles during design, effective detection and suppression systems, and specific measures to ensure that essential nuclear safety functions continue to be performed during a fire. It also required that the design of fire detection and extinguishing systems be based on a fire hazard assessment, which would provide formal documentation of the selection process normally performed by the fire protection system designers. The standard was revised in 1995, to include more direction for fire protection during plant operation, guidance for the review of fire protection at existing plants, and guidance for daily, weekly, monthly and annual inspections of the fire protection features of the plant.

3. Fire Hazard Assessment of CANDU 6

The N293 standard was first applied to the Wolsong 2/3/4 plant in Korea, which is essentially the same design as the earlier CANDU 6 plants, with upgrades based on operating experience current nuclear plant standards. More extensive requirements applying the principles of the standard were included in design documentation submitted to the regulator early in the project. This was the first CANDU plant to be built that employed a formally documented Fire Hazard Assessment during the design of the fire protection systems. The assessment method used was developed during an earlier CANDU conceptual design [1], and used a standard format for all areas of the plant. The plant was divided into "fire zones" for purposes of evaluation of the hazards, which coincide with structural components and fire barriers where possible, but could also consist of open space where propagation across the boundary would be assessed.

For each zone, the following aspects were listed:

a) fire zone location: room number and elevation,

b) systems and major components

c) combustibles and ignition sources

d) fire barriers and separation

e) fire detection and extinguishing systems

f) access for fire fighting

g) assessment of fire hazards.

Judgment and experience were used to determine the adequacy of the fire protection measures for each zone, and the possibility of propagation to other zones, with each consideration recorded in the Fire Hazard Assessment In the assessment of fire hazards, the primary consideration was whether the essential safety functions of shutdown, fuel cooling, containment of fission products, and control and monitoring of the shutdowr plant were maintained In this respect, performance of the essential nuclear safety functions by the two widely separated independent groups proved to be invaluable, avoiding the use of sophisticated analyses to defend fire barriers and individual components

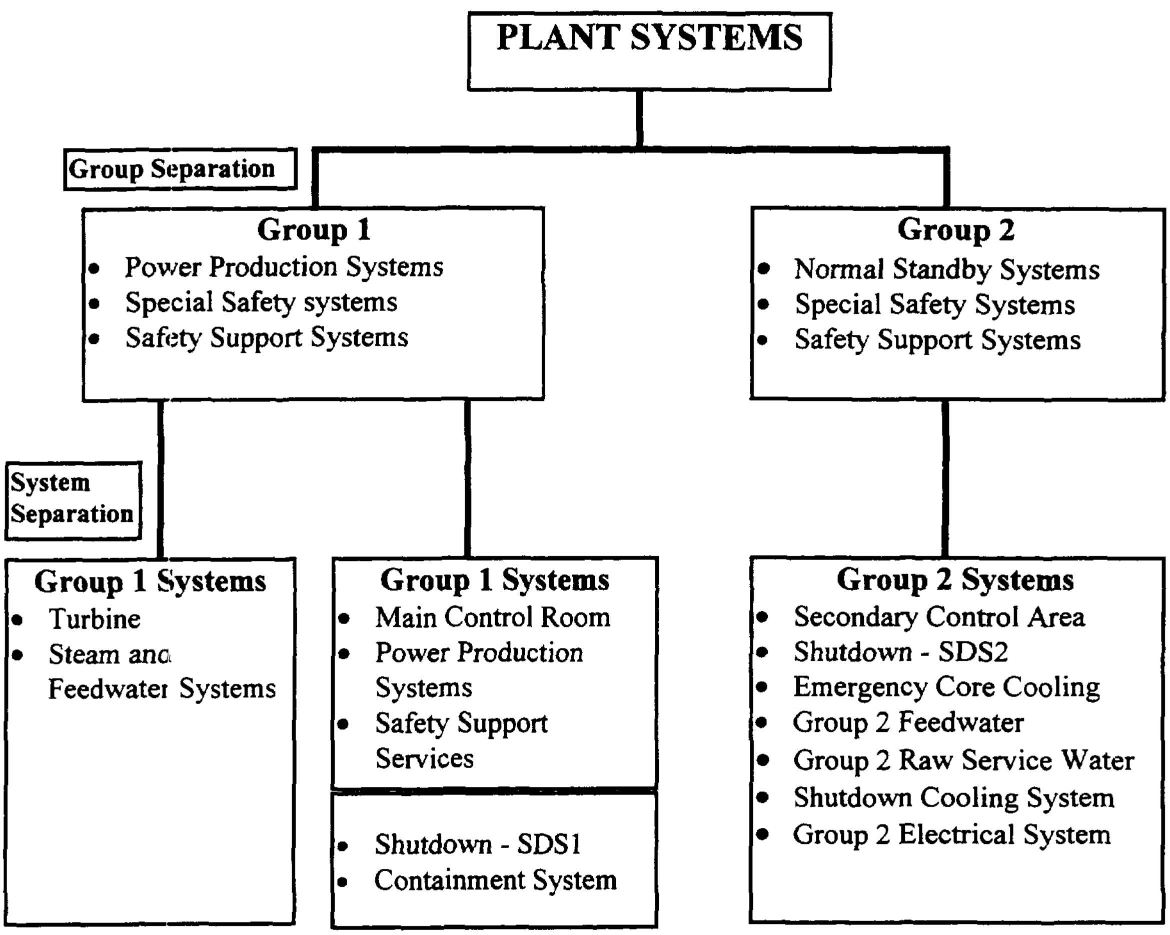

Figure 3. Grouping of Safety Related Systems for CANDU 9 Plants

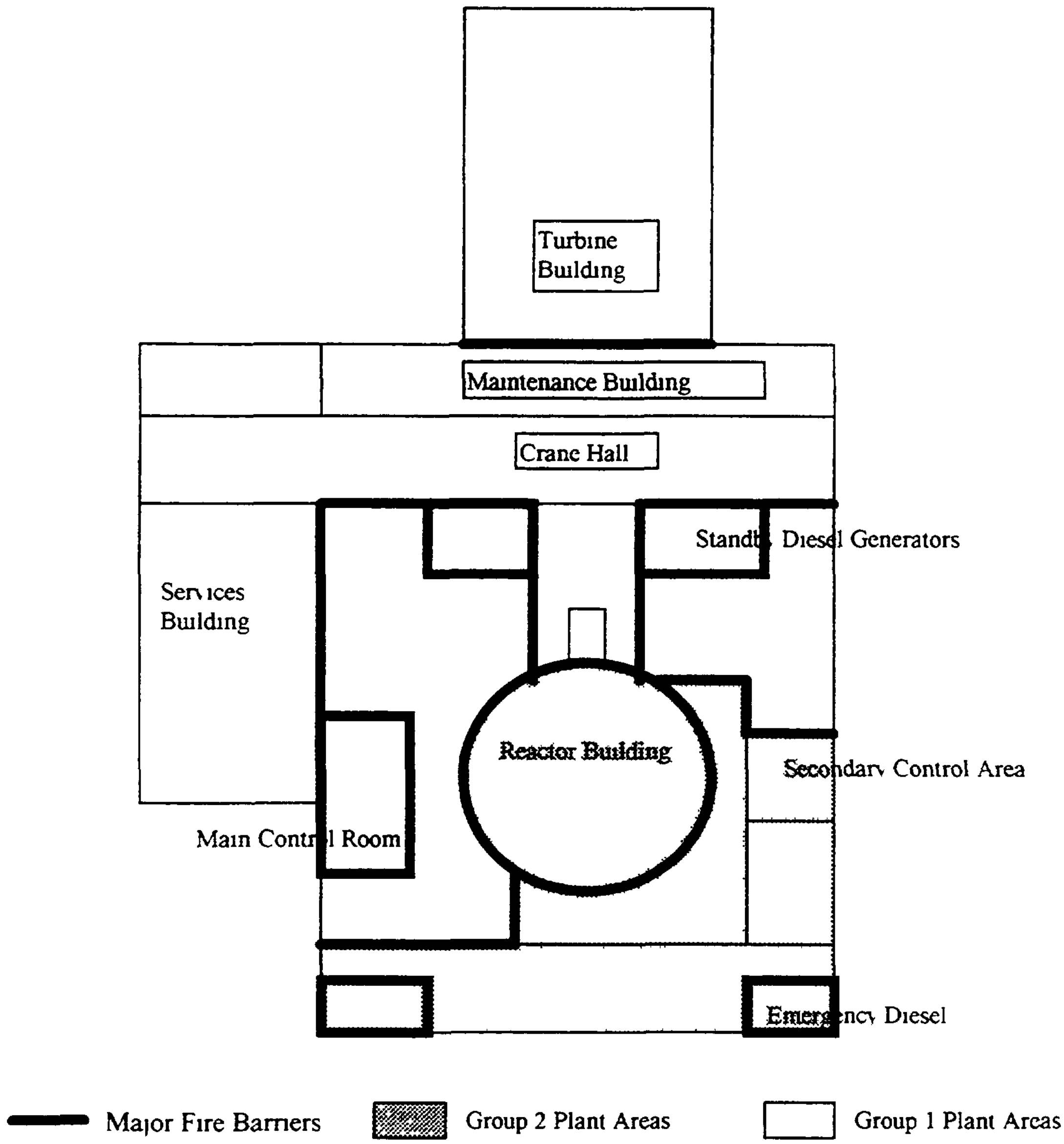

Figure 4 CANDU 9 Layout and Major Fire Barriers

4. Application of Fire Protection to CANDU 9

The most recent CANDU plant is the new CANDU 9 design, where current fire protection principles have been extensively incorporated during the conceptual design The plant layout builds on the two group separation principle developed in the CANDU 6 plants Figure 3 shows that the number and capability of the Group 2 systems has been increased considerably over that of previous designs Figure 4 shows the location of the Group 1 and Group 2 areas, and the major fire barriers between groups and within groups The layout is more compact to optimize the use of a site in a multiunit configuration, and to enable the safety related systems to be protected against site related events The layout also improves the separation of safety related equipment from fire hazards (e g the main standby generators and electrical buses are located within the reactor auxiliary building), the separation of redundant components within systems, and the minimization of combustible materials in the area of safety related systems This is backed up by well defined fire barriers, fire detection in all plant areas, and automatic fire suppression in all plant areas except those continuously occupied and in the reactor building A seismically qualified fire water supply is provided for the reactor building and the Group 2 area of the plant containing essential safety related equipment credited in the seismic event A comprehensive Fire Hazards Assessment and Probabilistic Safety Assessment for internal fires will be completed during the detailed design, providing extensive documentation for the plant operating staff for their fire protection program

5. Conclusion

The evolution of fire protection practices in CANDU plants has mirrored the development of fire protection principles and practices that has taken place internationally, and the CANDU design has benefited from the participation in, and guidance of, the IAEA initiatives for nuclear power plants. The formal Fire Hazard Assessments for current CANDU plants have contributed to the improvement of the fire protection design features, and provide extensive information to the operating plant as a basis for their fire protection program

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IAEA-SM-345/8

SAFETY IMPROVEMENTS MADE AT THE LOVIISA NUCLEAR POWER PLANT TO REDUCE FIRE RISKS ORIGINATING FROM THE TURBINE GENERATORS

XA9847530

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Abstract

Comprehensive upgrading measures have been completed for the Loviisa Nuclear Power Plant (modified VVER440/V213). These were carried out from the start of the design phase and during operation to ensure safe plant shutdown in the event of a large turbine generator oil fire. These modifications were made mainly on a deterministic basis according to specific risk studies and fire analyses. As part of the probabilistic safety assessment, a fire risk analysis was made that confirmed the importance of these upgrading measures. In fact, they should be considered as design basis modifications for all VVER440 plants.

1. INTRODUCTION

Operational experience has proved that disturbances of turbine generator systems often affect overall plant safety. Many disturbances have led to accidents causing accidents and extensive damage in turbine halls and in their vicinity. The most severe consequences have been loss of habitability of the control room, loss of residual heat removal and total loss of electrical power for a longer period.

The frequency of events originating from the turbine generator is much higher than initially estimated. The most serious consequences of turbine generator failures seem to be fires, which have a very high occurrence rate. For instance in VVER-440 plants, turbine generator damage can lead to loss of main feed water, emergency feed water and the primary circuit residual heat removal systems which are situated in the turbine hall.

This paper is based on experiences of the fire and risk analyses conducted by STUK and the PSA analysis conducted by IVO as well as the safety improvements implemented at Loviisa NPP. In this paper implemented improvements are presented and also new potential proposals are considered.

2. THE RISKS OF TURBINE GENERATOR SYSTEMS

In the turbine hall of a VVER-440 nuclear power plant the following safety-related systems are typically located:

- main feed water system
- emergency feed water system
- residual heat removal system
- steam lines including safety and isolation valves.

At the original VVER-440 plant, loss of these systems causes the loss of heat removal via the secondary circuit which is needed for the primary circuit cooling. Many improvements have been carried out to ensure heat removal from steam generators.

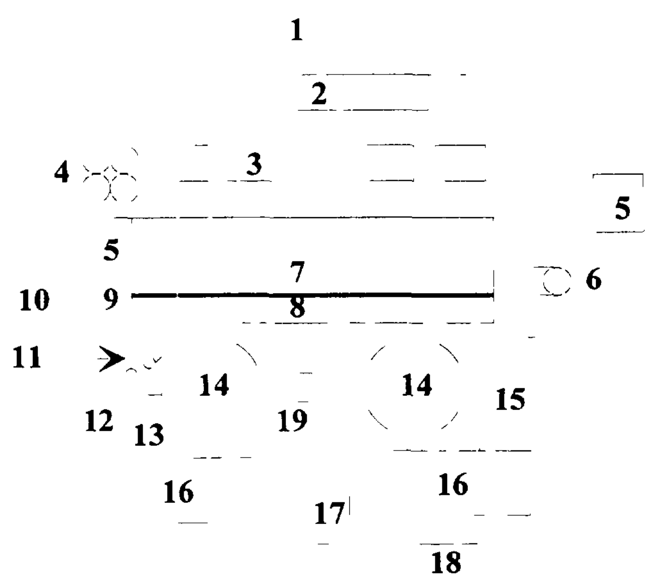
At Loviisa NPP (a modified VVER-440/V213 plant) buildings connected to the turbine hall are (fig. 1):

- control building
- sea water pumping plant of Loviisa-1
- demineralisation plant.

Especially the control building is important for plant safety. Electrical rooms (I&C, accumulators, switchgears) and cable rooms are situated inside the same building (fig. 2). The main steam and feed water pipes and valves are located on the top of the control building. In anticipated transients heat removal by the turbine condenser and the main feed water system may be lost. In this situation the residual heat can be removed from the primary circuit by the feed and bleed functions of the steam generators:

- feed by the available emergency feed water system
- bleed by the controlled main steam valve blow
- isolation of the main steam and main feed water lines to maintain the pressure and the water level in the steam generators.

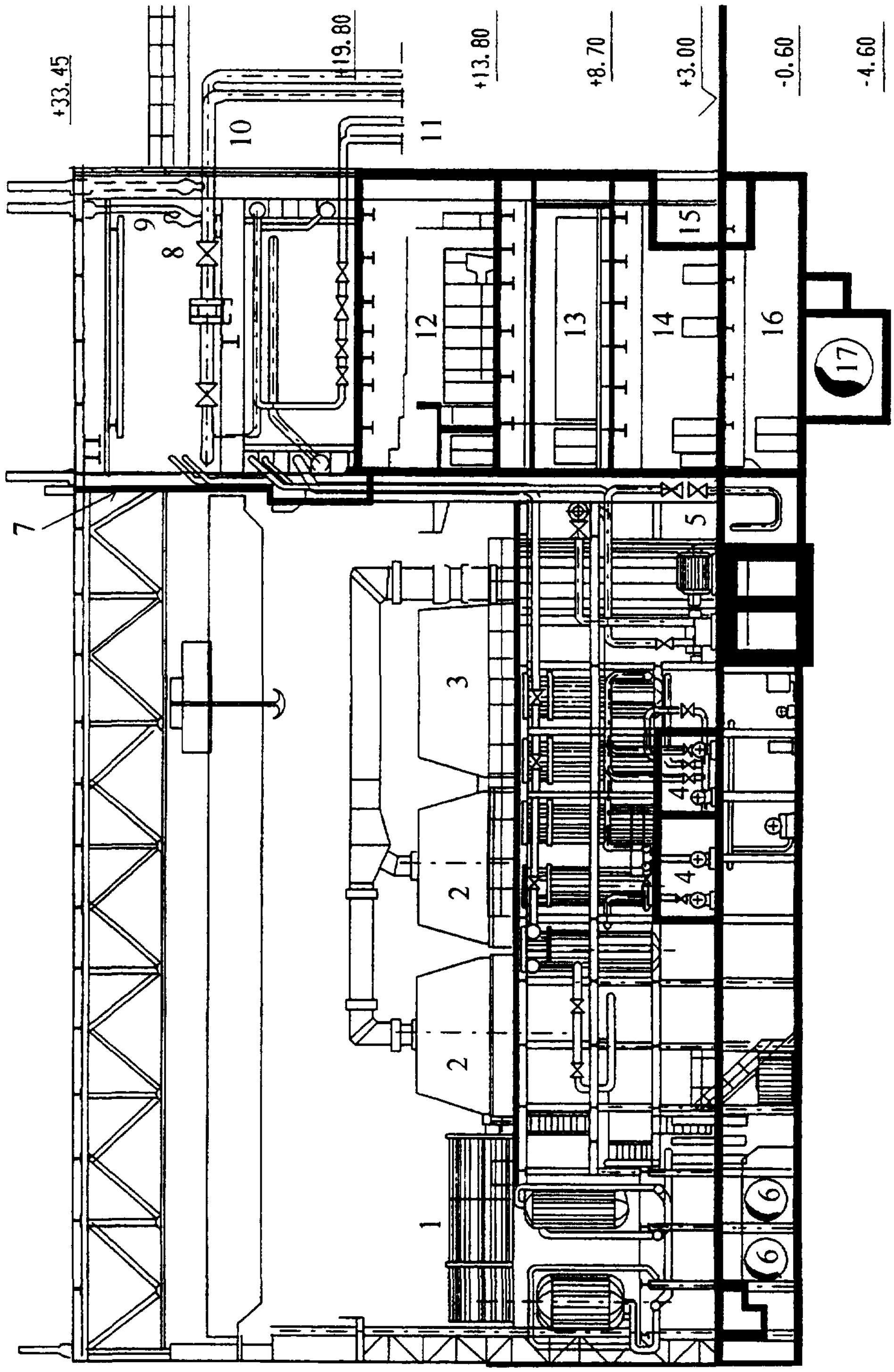
The equipment needed for these feed and bleed functions are vital for the plant's safety and they should be separated from the turbine hall.



1 400 kV switchyard
2 diesel building
3 main transformers
4 storage tanks of demineralizated water
5 sea water pumping plant
6 fire water pumping plant
7 turbine hall
8 control building
9 demineralization plant
10 office building
11 emergency feed water pumping plant and excess water tank area
12 laboratory building
13 roofed tank area
14 reactor building
15 service building
16 auxiliary building
17 social building
18 spent fuel storage
19 emergency water tank

Figure 1. Power plant area of Loviisa NPP. Thicker line between turbine hall and control room building is fire wall.

Almost all of the plant safety systems are cooled by the service water system which is located in the sea water pumping plant together with the turbine condenser cooling water pumps (big sea water pumps). Of these safety systems, the residual heat removal system for unit cold shut down and the conventional intermediate cooling system for cooling of the feed water and emergency feed water pumps are situated in the turbine hall. If the turbine condenser cooling is not operable, the conventional intermediate cooling system can also be cooled by service water pumps.



1	*generator*	*9*	*main relief valves*
2	*low pressure turbine*	*10*	*live steam lines*
3	*high pressure turbine*	*11*	*feed water lines*
4	*pumps of emergency feed water and residual heat removal systems*	*12*	*main control room*
		13	*rooms for instrumentation*
5	*feed water pumps*	*14*	*switch plant*
6	*sea water entry*	*15*	*house-load transformers*
7	*fire wall (also called B line wall)*	*16*	*cable rooms*
8	*fast acting isolation valves*	*17*	*sea water egress*

Figure 2. Cross-profile of the turbine building.

The most significant turbine generator failures are missiles ejected from turbine and fires which lead to severe consequences. Turbine generator failure modes, frequencies and risks have been studied more in the paper "Risks of turbine generators at VVER-440 nuclear power plants". In this connection it is important to notice that a large oil fire in the turbine hall is a potential risk for all safety-related systems situated in the turbine hall or in its vicinity and therefore a reason for serious concern.

3. IMPLEMENTED PLANT MODIFICATIONS

Deficiencies in many plant sections have been identified especially in the original lay-out design of VVER-440 plants. Insufficient fire compartmentation and component separation as well as poor fire resistance of structures can be considered as the biggest problems. To improve the deficiencies of basic design, many modifications have been already implemented starting from the design phase and early stages of plant operation. The most significant modifications have been:

- Installation of sprinkler systems to protect the turbine hall, as a whole area sprinklers for different levels and local sprinklers for oil systems.
- Fire insulation of the load bearing steel structures of the turbine hall and the control building.
- Installation of a fire valve to the governing oil line. The valve will close automatically when the fire detection system detects a fire in an HP turbine. The closure will trigger a turbine trip due to a pressure drop of the regulating oil.
- Adding the option of feeding water to the steam generators from the primary circuit make-up water pumps.
- Relocation of the generator hydrogen cooling system's hydrogen station and lines to a more safe place.
- Construction of a fire wall (F180) between the turbine hall and the control building (so called B line wall, fig. 2). The fire wall separates the steam and feed water lines and valves from the turbine hall into an individual fire compartment to protect them against turbine hall fires. The wall is constructed to withstand the potential collapsing of the roof. Also, smoke and heat removal hatches were added.
- Setting up a new back-up emergency feed water system. The system is totally independent of the feed water and emergency feed water systems in the turbine hall. The system's pumps are diesel-driven and they are able to supply water directly to the bottom of the steam generators (to four of six).
- The cooling and ventilation systems of the control building were improved (also against fires in the turbine hall).
- Relocation and compartmentation of safety-related cables and equipment in the turbine hall.
- Fire protection improvements at the governing oil system of the turbine by-pass valves (on the top of the control building) .
- Building up a new, separate fire water pumping station, which consists of three diesel driven water pumps, water tanks, supply lines and valve stations.
- Replacement of primary circuit safety valves with new type Power Operated Relief Valves (PORVs), which make the primary circuit feed and bleed function possible.
- Improvements in the generator's hydrogen cooling and sealing oil systems.

All the improvements mentioned above have been reported widely in STUK's quarterly reports, "Operation of Finnish nuclear power plants".

The latest improvements concern the hydrogen cooling and sealing oil systems, which have been modified in the annual maintenance outages in 1996 and 1997. The main aim has been to improve the operational safety of these systems and also to modify the systems to meet the current standards. One reason for the modifications was also the small hydrogen fire at unit two on 17 January, 1996. The fire resulted from a leaking liquid detector. The implemented modifications during the year 1996 were:

- changing of liquid detectors and respective valves
- potential equalisation of sealing oil system pipes.

During the 1997 maintenance outage a lot of changes were made in the hydrogen cooling system of Loviisa-1. Because of the changes the system had to be redesigned and built up almost from the beginning. Mostly due to the lack of time modifications were made only for one generator. The objectives of the changes were:

- to move the valve cabinet of the cooling system to adequate distance, as also required by new safety standards, from the generator's electric systems
- to simplify the system and thus enhance its operability by deleting unnecessary valves and pipes
- to replace the flanged joints of pipes and valves with welded joints where possible
- to facilitate nitrogen feeding to the generator during the hydrogen dryer's regeneration.

4. PROPOSALS TO DECREASE THE RISK RESULTING FROM TG FAILURES IN FUTURE

Proposals are divided into three groups. Division has been made by figuring out, how demanding the improvements are and how much the planning and the implementation would take time. The purpose has been to evoke discussion between the utility and the authority about possibilities to improve safety of turbine generator system. Most of the proposals are aimed to decrease the frequency of turbine generator failures or their consequences.

4.1. Short term improvements

Short term improvements concern the plant's instructions, tests and staff training. Despite their easy implementation, the improvements may be significant for the systems' operational safety. The most important subjects are:

- operational instructions
- instructions for disturbances and emergency situations
- tests and instructions for testing
- simulator training of operators

The applicability of the plant's instructions has been proven in the course of time. Mistakes and shortcomings in the instructions have been handled with the operation group manager.

The simulator training programs have not contained enough turbine generator events considering their frequencies and significance.

4.2. Medium term improvements

Medium term proposals concern mainly the construction and instrumentation of the turbine generator systems. Modifications can be performed within a rather short time period and with reasonable economical investments. The most important subjects for proposals are:

- vibration monitoring system
- hydrogen cooling system.

4.2.1. Vibration monitoring system

The length of time delays, the small quantity of measurement sensors and the lack of automatic turbine trip due to high vibration levels are the biggest drawbacks of the present-day vibration monitoring system. To improve the situation a modernisation project has already been started at the plant. It has been found out that modern monitoring systems with versatile and fast analysing resources are very good instruments for the surveillance of the turbine generators. However, the automatic trip function has not been included in the plans in this phase.

The attitude against the automatic trip function is not based on operational experience. Such a function is generally used all over the world. In Finland, Olkiluoto NPP has very good experience from it. Any problems, such as false trips, have not occurred. It is important to notice the enormous benefits of an automatic trip function: it is almost the only way to drive the turbine generator down quickly and safely in case a sudden unbalance due to the breakage of the rotating parts occurs.

4.2.2. Hydrogen cooling system

The hydrogen cooling system modifications have obviously improved the system's safety. The most significant weaknesses that were not covered by the modification plans are those connected to the hydrogen dryer, the analysers of purity and humidity and the lay-out of the components.

The problem of the silica gel dryer is its inefficiency. It is impossible to attain the recommended level of hydrogen humidity set by the generator's manufacturer by using the present dryer. Especially after the maintenance shut-down, the humidity may exceed the permitted value considerably. The key to the humidity problems would be to replace the dryer with a more efficient model.

The humidity of the hydrogen is controlled by laboratory analysis once a month. If the humidity increases for instance due a water leak from the hydrogen cooler into the generator, the observation may fail for a long time. Therefore an on-line humidity analyser should be installed in the cooling system.
The best alternative when safety is concerned is that the hydrogen dryer and valves are situated in a place resistant against fires and explosions, protected with leak monitoring and extinguishing systems. This has proved difficult to be achieved in practice. After all, the monitoring of the hydrogen leaks in the vicinity of the hydrogen dryer and the valves should be the target of the modifications.

4.3. Long term improvements

Long term improvement needs consist of very extensive changes in the systems, which will require further system and risk studies. A lot of effort has been put forward to find the areas in which large modifications would be reasonable for improving safety e.g.:

- improving operational safety of the oil systems
- protection of the safety-related systems situated in the turbine hall.

4.3.1. Oil systems

Subjects still unaddressed are:

- vibration monitoring of the oil pipes
- supervision of the oil pipes
- drainage areas of the oil-leaks
- lay-out of systems pipes and components.

4.3.2. Protection of the safety-related systems

Safety-related components situated in the turbine hall have been planned to be protected by using sprinkler systems and barriers. However, some components are still without adequate protection. It is important that the protection of the components will be examined under the light of probabilistic risk analysis.

5. CONCLUSIONS

Many improvements have been carried out at Loviisa NPP (chapter 3). The improvements have decreased the risk of turbine generator failures considerably. The independent back-up emergency feed water system, the fire separation wall of the turbine hall and other fire safety improvements decrease the total core damage risk at Loviisa NPP roughly by a factor 100 according to the present PSA model of the plant. The risk of turbine hall fires seems to be on the same scale of magnitude as the influences of many other initiating events. This argument is supported by a fire risk analysis, which has been completed by the utility and delivered to STUK for inspection. A fire in the turbine hall, especially when resulting from a turbine generator mechanical breaking, is still one of the most considerable contributors to core damage. There are still some areas where changes are necessary and cost-efficient. Such areas would be e.g. the turbine generator vibration monitoring, the generator hydrogen cooling, the short term proposals indicated above and the protection of safety-related systems.
At some other VVER-440 plants the situation is much worse. First of all, the lay-out is usually weaker than at Loviisa. The most important weakness is the unfavourable location of turbine generators. Lay-out modifications are mostly impossible to carry out, but with some improvements the situation could improve significantly. The most effective improvements reducing core damage risk are:

- building up an independent back up emergency feed water system
- construction of a fire wall (F180) between the turbine hall and the control building and valve gallery
- insulation of load bearing steel structures of the turbine hall and the steam and feed water valve gallery

- installation of sprinkler systems to protect the turbine hall
- building up a modern fire water pumping station.

At least the first two improvements are extremely important for reactor residual heat removal and these systems can be recommended for all VVER-440 plants. Without improvements the core damage risk may be two decades higher than in Loviisa and the significance of the turbine hall is emphasised accordingly. After these basic design improvements have been made, safety can be further improved by utilising the proposals presented in chapter 4.

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Invited Paper

UPGRADING OF FIRE SAFETY IN INDIAN NUCLEAR POWER PLANTS

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Abstract

Indian nuclear power programme started with the installation of 2 nos. of Boiling Water Reactor (BWR) at Tarapur (TAPS I & II) of 210 MWe each commissioned in the year 1969. The Pressurized Heavy Water Reactor (PHWR) programme in the country started with the installation of 2x220 MWe stations at Rawatbhatta near Kota (RAPS I & II) in the State of Rajasthan in the sixties. At the present moment, the country has 10 stations in operation. Construction is going on for 4 more units of 220 MWe whereas work on two more 500 MWe units is going to start soon. Fire safety systems for the earlier units were engineered as per the state-of-art knowledge available then. However, there have been several developments since then in the understanding of the subject of Fire Safety. New engineering information by way of standards and codes from various national as well as international organizations, materials of construction for cables, fire barriers etc, construction practices, surveillance requirements, detection systems have since been available in the market. Feed back available from several fire incidences within as well as outside the country have taught number of lessons. The engineering design as well as the installation of the existing fire safety system at the plants need to be reviewed with reference to these in order to ensure better protection against fire incidences.

The need for review of fire protection systems in the Indian nuclear power plants has also been felt since long almost after Brown's Ferry fire in 1975 itself. Task forces consisting of fire experts, systems design engineers, O&M personnel as well as the Fire Protection engineers at the plant were constituted for each plant to review the existing fire safety provisions in details and highlight the upgradation needed for meeting the latest requirements as per the national as well as international practices. The upgradation at one of the two units of a plant was done during rehabilitation after a fire incidence on the unit. The fire protection system design for the 4 units under construction as well as the 500 MWe units yet to be taken up for construction was already reviewed to meet the current requirements/standards.

The recommendations made by three such task forces for the three plants are proposed to be reviewed in this paper. The paper also highlights the recommendations to be implemented immediately as well as on long-term basis over a period of time.

1.0. INTRODUCTION

Electricity generation through the nuclear route is a very important option for meeting the energy requirement of any country. Although presently in India, nuclear energy contributes only around 2% of the total electricity generated in the country, in view of the environmental problems associated with the use of fossil fuels as well as limited availability of these in the immediate future, nuclear energy becomes a very important alternative for producing electric energy.

Indian nuclear power programme started with the installation of 2 nos. of Boiling Water Reactors (BWRs) at Tarapur (TAPS I & II) of 210 MWe capacity each in the year 1969. The programme is based on Pressurised Heavy Water Reactors (PHWRs). The first two PHWR units of 220 MWe capacity each were installed at Rawatbhatta near Kota (RAPS I & II) in the state of Rajasthan in the year 1973. At the present moment, the country has 10 stations in operation. Construction is in progress at 4 more units of 220 MWe each. Work on 2 units of 500 MWe each is going to start soon. Details of the Indian nuclear power programme is given in annexure-I.

Fire safety systems in the units installed initially were engineered as per the state-of-art knowledge available then. There have been continuous advancements in the field of fire safety engineering and several developments in the understanding of the subject. It has therefore been necessary to continuously review the engineering design as well as installation of the existing fire safety systems at the plants and upgrade these with reference to the knowledge available now in order to ensure better protection against fire incidents.

The need for review of fire protection systems in the Indian nuclear power plants (NPPs) also has been felt since long almost after the Brown's Ferry fire in 1975 itself. This incidence caused an almost global awareness to look into the subject of fire safety afresh. New engineering information by way of standards and codes from various national as well as international organisations, better materials of construction for cables, fire barriers etc., construction practices, surveillance requirements, detection systems have since been available in the market. Feed back available from several fire incidents within as well as outside the country has taught lessons to learn.

The exercise for reviewing the existing fire protection system engineering to identify the weaknesses in the existing systems at the country's NPPs with a view to upgrade the fire safety was taken up in a very systematic manner. Details of these studies are discussed in the ensuing paragraphs.

2.0. PHILOSOPHY OF FIRE SAFETY REVIEW

The fire safety systems for the plants were designed to be in general conformity with the National Fire Protection Association (NFPA) Standards of USA, being most commonly followed globally. Fire Protection Safety Guide 50-SG-D2 issued by IAEA, fire protection standards 1.120 of U.S. Nuclear Regulatory Commission (which was later on superseded), Canadian, French & German Standards gave detailed guidelines and latest requirements for the fire protection system design. Atomic Energy Regulatory Board, which is the regulatory body for the Indian nuclear installations also issued Fire Protection Standards and the design safety guide is under issue.

The review of the fire safety provided at the various NPPs was planned to be conducted through knowledgeable persons having sufficient knowledge in the field of fire safety, engineering of the systems provided at the plant, knowledge about the codes, knowledge about electrical as well as control and instrumentation systems designed for the plants etc. Dedicated task forces were constituted for the individual plants who were required to look into the detailed provisions at the respective plants, analyse, evaluate and then identify weaknesses in the specific areas. The task forces were also expected to carry out hazard analysis for the critical plant areas.

3.0. PHILOSOPHY OF FIRE SAFETY UPGRADATION

After reviewing the fire protection systems provided at the operating stations, the weaknesses are identified. The reports produced by the task forces are formally discussed in the respective Station Operations Review Committees (SORC) appointed for each station. Once the recommendations are finalised, these are then taken up for implementation.

Depending upon the constraints in the implementation, the action plan is decided to execute these upgrading activities on short term or long term basis. The constraints could be by way of working out engineering details and getting these designs cleared through the regulatory authority, procurement of materials, need for a station shutdown for short or long duration etc.

Although it will be desirable to have the fire protection system design as well as the plant layout to meet the latest standards, practically it is not possible. This has been experienced even globally also. US N.R.C. in May 1976 issued Branch Technical Position, Auxiliary and Power Conversion Systems Branch 9.5-1 (BTP A - PCSB 9.5-1) "Guidelines for Fire Protection of Nuclear Power Plants" which applied to plants licensed for construction after July 1, 1979. In order to establish an acceptable level of fire protection at the older, operating plants without having a significant impact on plant design, construction or operation, the NRC modified the guidelines in the original BTP in September 1976 and issued Appendix A to BTP 9.5-1 "Guidelines for Fire Protection for Nuclear Plants Docketed prior to July 1, 1976".

In India, although no codes have been issued for fire protection acceptable levels in respect of plants constructed as per older designs exclusively, the philosophy of defence-in-depth was applied to each case on its merit.

For the new plants, adequate provisions have been made in the designs to meet the requirements of the available national and international codes, some of which are as under:-

IAEA Safety Series No. 50-SG-D2	- Fire Protection in Nuclear Power Plants.
NRC Guide 1.75	- Physical independence of Systems.
IEEE 384	- Criteria for Independence of Class IE equipment and circuits.
NRC 10 CFR 50 Appendix R	- Fire Protection
NFPA Codes	
AERB/S/IRSD-I	- Standard for Fire Protection Systems of Nuclear Facilities, issued by Atomic Energy Regulatory Board.

Fire Protection Codes & Standards issued by Bureau of Indian Standards.

The Regulatory authority is also bringing out Safety Guide No. AERB/SG/D-4 on "Protection Against Fire in Nuclear Power Plants" shortly.

As in the case of NRC, for upgrading at Indian plants also in respect of 3 hours fire barriers everywhere, a decision was taken that wherever automatic suppression and detection systems have been provided, a 1 hour rated barriers can be considered an equivalent level of protection to that provided by a 3 hour barrier without suppression or detection between redundant trains/equipment.

4.0. UPGRADATION OF FIRE SAFETY IN THE OPERATING NPPs

As has been discussed earlier, in order to review the fire safety measures provided in the existing operating plants and then identify the weaknesses in order to upgrade the fire safety systems, task groups were constituted for each of the operating plants. The task groups after reviewing the systems and combustibles present in the various areas in details, and carrying out a fire hazard analysis, put forward recommendations which are discussed in the succeeding para.

4.1. Tarapur Atomic Power Station

TAPS being the oldest station, weaknesses were identified by the task force. The reports of the task force is still under finalisation. However, the following recommendations need attention:-

4.1.1. Recommendations

4.1.1.1. Cables

1. Fire retardant coating with a minimum fire rating of 30 minutes needs to be applied to the horizontal as well as vertical cable runs. All horizontal runs of cables to be coated for 2 meters length at the interval of 6 metres. Vertical cable runs could be coated on their entire length between floor and ceiling.

2. For important safety related equipment, cables should be segregated, if not already done so.

3. As far as possible, all future procurement of instrumentation and electrical cables should be of Fire Retardant Low Smoke (FRLS) type.

4.1.1.2. Ventilation

1. Wherever not provided, ventilation fans for the main plant buildings should be interlocked with fire detection system to trip on detection of smoke/flame.

2. Smoke evacuation systems should be provided in cable spreading room. Proper procedures need to be written for exhausting smoke from the Turbine building through roof exhausters.

4.1.1.3. Fire Barriers

Provision of fire barriers need to be reviewed particularly in respect of the safety related equipments. Wherever not provided, Fire barrier of appropriate fire ratings need to be installed.

4.1.1.4. Fire Detection & Suppression Systems

The fire detection system provided is of old design. Recommendations have been made to consider upgrading the existing system with the later versions by installing addressable detectors. Also review of existing detector coverage with diverse type detectors has been suggested. Provision of dykes around the oil tanks needs to be made. Provision of 10 SCBA sets is also recommended for the control room.

4.1.1.5. Augmentation

Diesel generating set for Station Blackout (SBO) condition would supply emergency power to the station equipment.

Fire safety in reactor instrument vault may be reviewed and upgraded.

4.1.2. Upgradation Plan

Some of the recommendations can be readily implemented and action has already been initiated. Some recommendations need engineering review and material procurement after which these can be implemented. For some recommendations, alternative measures such as increased surveillance, augmentation of passive measures, administrative controls etc. have been considered. However, there are a few recommendations which need in-depth engineering review and these have been planned on a long term basis.

4.2. Rajasthan Atomic Power Station (RAPS)

4.2.1. Recommendations

4.2.1.1. Cables

The following needs to be looked into for the cables and cable runs throughout the plant:

1. All horizontal runs of cables to be coated with 30 minutes rating fire retardent paint for 2 meter length at the interval of 6 meters and vertical runs in entire length from floor to ceiling.

2. All trunk routes of cable trays should have manual fire water sprinkler system.

3. Cable spreading room below 3.3 KV switch gear should be enclosed by a wall and be provided with sprinkler system. This room should be provided with separate duct-fan type smoke evacuation system.

4. Cable-bridge area to be covered with G.I. sheets and manual/remote sprinkler system to be installed in this area.

5. All cable-trays penetrations to be sealed with one hour rating fire barriers.

6. All safety related cable routes to be divided on the basis of Group-1 and Group-2 philosophy.

4.2.1.2. Ventilation Systems

1. R/B exhaust ventilation should have a spare bank of prefilter and absolute filter common to both the units in some other area.

2. One remote/manual operated damper to be provided in R/B exhaust duct at 1263′ floor for purging smoke from boiler room.

3. All presently installed fusible link type fire dampers to be replaced with air operated type fire damper of one hour fire rating for formation of fire zones.

4. Control room, control equipment and MCC room should have a separate fan duct type smoke evacuation system and control room make-up air should be taken from T/B ventilation duct instead of turbine building ventilation duct.

5. Control room and control equipment room R-1 and R-2 should have isolation damper in supply and return duct to contain the smoke in the affected area.

4.2.1.3. Fire Dtection and Protection System

1. Presently installed fire detection and alarm system to be replaced with addressable type fire detection and alarm system having total 209 address points of various types for each unit.

2. Areas outside operating island should also have a fire detection and alarm system having total of 100 smoke detectors.

3. Main oil tank of turbine should be provided with CO_2 dousing system inside tank and gravity oil draining system to drain MOT oil to outside turbine building. If it is not feasible, then provide draining into dirty oil purifier tank inside T/B.

4. Hydrogen leak detectors to be installed in turbine building near generator and battery room.

5. To meet the station black-out condition due to fire, RAPS-1&2 fire water header may be connected to RAPP-3&4 fire water header.

6. Fire hydrants outside operating island to be connected to fire water system.

4.2.1.4. Fire Barriers

1. All doors inside reactor building, control room, battery room, DG room, MG room should be tested for fire-rating to meet the fire rating requirement of one hour.

4.2.2. Upgradation Plan

The implementation of the above is in progress during the current capital maintenance outage of the RAPS-Unit 2 alongwith the coolant channel replacement work. The work in Unit-I will be taken up at the earliest opportunity.

4.3. Madras Atomic Power Station

4.3.1. Recommendations

4.3.1.1. Cables

1. Fire retardant coatings of minimum 1/2 hour rating needs to be provided for a length of 2 metres after every 6 metres and on the vertical runs between the floor and ceiling. About 1 metre length of the cable needs to be coated at the entry to the panel for all cables entering the panels.

2. Segregation of important cables supplying to safety related cables need to be reviewed and cables segregated wherever required.

3. Fire protection on cables running close to high energy process system lines need to be reviewed.

4. Redundancy in power supplies to important loads need to be reviewed for supply from both the units.

4.3.1.2. Fire Detection & Protection Systems

Adequate physical segregation/installation of barriers between important safety related loads as needed.

4.3.1.3. Fire Barriers

1. Review of fire stops/brakes in the cable tunnel and corrective measures as appropriate.

2. Dykes/isolation for oil tanks.

3. Review of fire barrier/physical separation between important safety related equipment and corrective measures as required.

4. Improvement on some of the fire barriers provided already.

4.3.1.4. Ventilation

1. Fans for ventilation systems should be relocated to direct the exhaust in a proper manner.

4.3.1.5. House Keeping

1. Oil drums stored in SB near MM shop are to be shifted outside.

4.3.1.6. Supplementary Control Room

1. The supplementary control room as already recommended by MAPS should be provided.

4.3.1.7. Redundancy

1. The field flashing unit is common for both MG1 & 2. Independent field flashing units should be provided for each MG set.

It may be noted that the above are only some of the major recommendations. Wherever further weaknesses are noted during any stage, the same should be upgraded.

4.4. Narora Atomic Power Station (NAPS)

NAPS Unit I had a major fire on March 31, 1993. The fire resulting from failure of LP turbine blades started from the Turbine hall and spread upto control equipment room through ineffective fire barriers damaging several cables and ultimately resulting into a station blackout condition. The reactor was shutdown manually. Fortunately there was neither any damage to the reactor, nor spread of radioactivity nor death of any human being due to the incident.

Extensive modifications were carried out in the NAPS-I after the fire incidence during its rehabilitation.

4.4.1. Passive Fire Protection

a) Segregation of power and control cables of safety related equipments which are required to be available during and after fire.

b) Some of the local control panels are relocated to the easily accessible areas during fire.

c) LTG panels for both units have been physically separated.

d) Hydrogen charging station is relocated outside of the Turbine building. Hydrogen leak detection system has been introduced.

e) The Fire Water System in nuclear power plants performs additional duty as heat sink. During station blackout fire water is injected to steam generators through manually operated valves. These valves were relocated from inside reactor building to the outside to provide better access during emergency.

4.4.2. Fire Isolation

a) Ventilation system has been reviewed from the point of view of fire spread and isolation of area under fire. It is retrofitted with about 90 fire rated dampers in several strategic areas like electrical equipment rooms, control room, control equipment room etc. to isolate in the event of fire. Fire dampers of minimum one and half hour rating as per UL-555 in accordance with NFPA-90A and IAEA Safety Guide 50-SG-D2 are considered.

b) All cable penetrations are reviewed and upgraded with fire rated cable penetrations and fire stops. Additional fire rated stops have been installed.

c) Additional Fire barriers have been considered as permitted by the available space.

4.4.3. Minimising Combustibles

a) For safety related service the use of Fire Survival cables is considered. The other cables are upgraded to Fire Resistant Low Smoke (FRLS) type. All Control and power cables have been given fire retardant treatment.

b) Provision to drain the main oil tank in case of fire to an outside storage tank to minimize the fire load is considered.

4.4.4. Elimination of ignition source

Oil collection trays are provided beneath the probable leakage points in the oil piping. Local barriers to separate oil piping from cables and hot piping are provided.

4.4.5. Active Fire Protection

a) Existing sprinkler system is extended to cover the seal oil piping and turbine bearings. Additional sprinklers are planned for areas having higher cable concentrations like Cable Bridge, PHT passages and Mazzanine floor passage.

b) Addition of microprocessor based detection system having individually addressable detectors in cross zoning to ensure earliest warning of fire and minimization of false alarms. This also incorporates the controls for actuation of fire dampers & tripping of exhaust and ventilation system. Multiple fire detection systems have been considered for critical areas.

4.4.6. Control Room Habitability

Special emphasis has been given to the control room habitability during emergency. Operators are trained to use mask air stations, breathing apparatus, portable smoke exhausters and emergency lights which are provided along with portable Halon extinguishers. An alternate path is provided for fresh air intake for control room ventilation during emergency. Additional control for stopping and starting of smoke exhausters and AHU's feeding to control room is provided in control room. Gravity closing doors are considered between control equipment room and control room to facilitate their opening only on requirement.

4.4.7. Fire Protection Management

To improve the fire protection management, following provisions are considered.

1. A regular fire prevention plan is envisaged to take care of regular inspection and maintenance of fire barriers, fire sealing and fire retardant coatings.

2. Marking of Escape routes and emergency exits to show the way out for safety of personnel.

3. A Fire Fighting Preparedness Plan is envisaged for every plant area. This involves fire drills and training to the operators, house keeping and record of the safety procedures.

4. Pre fire plan for training and actual fire fighting.

5. Inviting constructive safety suggestions from plant personnel. This helps to maintain safety awareness, reduce combustibles and improve fire safety.

6. Upgradation of manual fire fighting crew and equipment.

7. Training of fire fighting crew to identify the plant areas, the inventory of combustibles and their evacuation requirements.

4.4.8. Implementation in Unit-2

Most of the above modifications have been carried out in NAPS Unit-2 also.

5.0. UPGRADING THE FIRE PROTECTION SYSTEMS

Implementation of upgrading efforts posed its own share of problems. Some of the problem areas were as under:-

1. Lack of space available in the plant for rerouting or providing physical separation.

2. Difficulty in segregating cables for various voltages and belonging to separate trains.

3. Lack of standardisation in the fire resistant properties for cables.

4. Availability of adequately fire rated fire barriers.

5. Availability of fire rated fire doors, dampers etc.

6. Economic considerations

Lack of space for physical separation of cables/ equipment was a problem that could not be easily solved in the older plants. Also introduction of physical barriers/partitions of appropriate fire rating could mean restriction of access during operation and maintenance of the equipment.

Lot of development efforts were put in to standardise on fire resistance coatings for cables, fire barriers of suitable fire ratings, good quality cables of appropriate low smoke as well as fire survival quality. Experiments were conducted in-house at our plants as well as at some of the established national laboratories like Central Building Research Institute (CBRI) at Roorkee. These efforts paid dividends and we were able to get the desired quality of cable coatings and barriers.

On the cables already laid out in thick bunches, installation of fire barriers at area penetrations also posed challenges and installation procedures had to be developed to suit each penetration.

Considerable difficulty is being experienced in the availability of adequately fire rated doors and dampers. Testing for the fire resistance quality to meet international standards e.g. ASTM E-152/NFPA 252 for doors or UL 555 for dampers has not been easy always. Doors and dampers tend to become very heavy in weight and pose difficulties in installation and operation.

Although no compromise on economic considerations for fire safety can be made, the expenses involved do become an important criterion in carrying out the upgrading programmes. Again here also, this is not only our experience but true globally. As such the upgrading programmes need to be reviewed for fire safety enhancement vis-a-vis economics involved.

6.0. CONCLUSION

To conclude

1. Indian nuclear power programme has attained maturity in all areas.

2. Fire Safety in the older plants has been reviewed through constitution of task forces. The recommendation of these task forces are reviewed from regulatory point of view.

3. Difficulties have been experienced in the availability of adequately fire rated equipment for the plants.

4. Upgrading has been done for the fire safety at the operating plants consistent with the defence-in-depth philosophy.

5. Review of fire safety system design is an ongoing process in our nuclear programme.

6. Fire Safety in the new plants has been provided as per the current international practices.

7. Fire Safety in the Indian nuclear power plants is given very high priority.

Annexure-1

NUCLEAR POWER — THE INDIAN PROGRAMME

Units	Installed capacity [MW(e)]	Year of commissioning and/or comm. operation	Remarks
In operation			
TAPS 1 and 2	2 x 160 (320)	1969	Units derated to 160 MWe because of the downgrading of some equipment.
RAPS 1 and 2	2 x 220 (300)	1973/1981	RAPS-1 derated to 100 MWe and RAPS-2 to 200 MWe.
MAPS 1 and 2	2 x 170 (340)	1984/1986	MAPS-1 & 2 derated to 170 MWe.
NAPP 1	220	1991	NAPS - 1 derated to 220 MWe.
NAPP 2	220	1992	
KAPP 1	220	1993	
KAPP 2	220	1994	
Under Construction			
KAIGA 1 and 2	2 x 220	1998/1999	
RAPS 3 and 4	2 x 220	1998/1999	
Planned			
2 units x 500	2 x 500	2003	Infrastructure already developed at Tarapur Site.
4 units x 220	4 x 220	2003	
2 units x 1000	2 x 1000		
4 units x 500	4 x 500		

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UPDATING OF THE FIRE FIGHTING SYSTEMS AND ORGANIZATION AT THE EMBALSE NUCLEAR POWER PLANT, ARGENTINA

XA9847532

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Presented by J.R. Martorelli

Abstract

A brief description is given of the updating carried out at the Embalse NPP after commissioning, covering the station fire equivalent loads, the station weak points from the fire point of view, the possible upgrading of systems or technological improvements, early alarm and automatic actions, organizations, education and training, and drills.

1. INTRODUCTION

Embalse NPP, located in the center of Argentina, is a CANDU — design plant of 648 MWe, commissioned in 1983.

It belongs to that design generation of the seventies, that's to say, it is more than 25 years old. It means that some upgrading was necessary to be done after commissioning; learnings from Browns Ferry and Three Mile Islands arrived too late for this projcet.

On the other hand, world background of recent years showed that fire hazard in nuclear power plants were greater and more serious than estimated in early days.

2. FIRE ACCIDENTS CHARACTERISTICS

It is known that origin and causes of fire accidents could be of two main sorts: technologial (due to device, equipment or system failure, such as: electric short-circuits or oil spillage on hot surfaces) or human (responsible of fire initiation because of a wrong work or responsible due to a lack of skill or poor preparedness to cope with the fire).

Probability of fire occurrence because technological causes can be lowered by corrective and preventive maintenance, routinary tests, automatic alarming and/or actions, systems upgrading, etc.

Human failures can be reduced by training, practice and education.

3. MAIN AREAS FOR IMPROVEMENT

Fire hazard in Embalse NPS was analyzed and seven main areas for improvements were found: a) Station fire equivalent loads. b) Station weak points from fire point of view. c) Possible systems upgrading or technological improvements. d) Early alarming and automatic actions. e) Organization. f) Education and training. g) Drills.

4. STATION FIRE EQUIVALENT LOADS

First of all, housekeeping was emphasized. No unnecessary materials were allowed to remain into the main buildings, that is: reactor bldg., service bldg., turbine bldg., auxiliary bay (electrical distribution area), and Diesels bldg.

Building separation was also emphasized, e.g. oil supply facility was moved out of turbine building to the oil storage house, specially built for this purpose. The same was done for inflammable compressed gases, such us hydrogen, acetylene, etc.

Wood, plywood, cardboard, plastics and paper were reduced as much as possible.

As a general rule, a very estrict control was established for inflammable materials handling.

5. STATION WEAK POINTS

A strong search took place at the plant looking for weak points in order to minimize the probability of fire.

On this field, even slight oil and fuel spillage were eliminated, dust and dirt cleaning got special attention and open flames were forbidden unless they were essential. Also smoking was restricted to a few areas where it is permitted.

Chemical fire inhibitors were used where possible, particularly in warehouse for stocked material and low inflammable solvents are in force for cleaning and degreasing. Essential inflammable chemicals are authorized only for particular applications and in limited quantities. PVC cables (not fire resistant) were protected with an inhibitor covering. Penetrations sealing got also special attention.

Places where fire is likely to occur a redundant fire detection system was installed, such as explosive gases detection for hydrogen in generator area and fuel vapors in Diesels area. These systems are in addition to the main fire detection system.

Besides, console of fire detection system was interconnected with security system console, so there is a permanent watching all over the plant by means of TV cameras, explosive gases detectors and smoke detectors. It is necessary to point out that security system was installed shortly after the station was built and commissioned. Security officers also overhaul the plant, in addition to the operations group, looking for abnormal or dangerous conditions, such as fire.

6. TECHNICAL IMPROVEMENTS

Several improvements were incorporated to the original design and some others are now in execution, while a bunch of them are still in project.

Improvements already done are, for example: a) The enlarging of security CCTV systen, with cameras in reactor and service buildings. b) Automatic shut down of ventilation system, dampeı closing and/or Halon flooding as a result of fire detection system activation. c) Halon protection for main and secondary control room, computer room, cable distribution room, equipment (signals distribution) room and emergency power supply room was also added to the original design. d) Manual high expansion foam and/or light water system installed in reactor building gives a better protection, not present in the original design. e) In the same way, a fire resistant screen was installed between normal and emergency water feed pumps. f) Interconnections carried out between fire and process water systems increase reliability of hydrants system. g) Several four inches couplings installed on the fire water system allows to connect the fire truck to this system, in order to use the fire truck as a booster pump to increase pressure from 8 kg/cm^2 to 40 kg/cm^2, this make possible to reach the plant highest point with the water jet. h) Separated warehouse for contaminated material and equipment. i) Halon protected bunker for essential plant documentation. j) Mock-up area for fire fighting practice.

Upgradings now in execution could be: a) CCTV System enlarging (turbine building). b) New protective equipment for fire brigade (garment for high temperature, newly design hard hats, etc). c) Emergency stock room with essential material and equipment foreseen for fire accidents. d) Replacement of methane gas system by P-10 gas (Argon-methane) in radiation monitors.

Some projected upgradings which are still one the desk could be: a) Halon replacement, b) Water mist fire protection for primary heat transport system main pumps in reactor building, c) High expansion foam system for turbine oil tank in addition to the actual dry-pipe system. d) High expansion foam protection for cable duct in reactor building. e) Doors replacement by stronger ones from fire point of view.

7. EARLY ALARMING AND AUTOMATIC ACTIONS

Permanent survey by means of CCTV allows image change (or movement) automatic detection and alarming. This will be used first in turbine building monitoring for fire hazard.

If this experience is successful, it will be expanded to Diesels building.

In Diesels building, Halon flooding manual system is being studied to be automatic or, at least, hand operated from main control room.

8. ORGANIZATION

Today, Embalse NPP has six fire brigades composed of operators on shift. They demonstrated to be efficient and fast.

Additional fire fighting support organization was formed with personnel which belongs to the radioprotection and safety department, personnel on call and personnel which belongs to National Gendarmerie, the armed force which protects the nuclear station, which is located beside the plant.

On the other hand, a coordination was established with external firemen 35 km around the station for the case of a fire at the plant. It was practically checked that in this situation at least 10 fire trucks could be at the place in about 20 minutes.

9. TRAINING AND EDUCATION

Fire brigades had an initual training and then a permanent retraining twice a year. This includes skill upgrading at the mock-up yard for fire fighting practice.

All plant personnel (including women) receive a theoretical and practical retraining on fire fighting each four years.

External firemen are retrained each four years on the basic knowledge of radiological protection and station characteristics in order to be able to behave safely in case of a fire at the plant.

Additionally, it is considered essential to accomplish with a permanent education of personnel on the way of a safe behavior in all performed activities.

10. DRILLS

There are at least six fire fighting drills a year, that means an average of once a year for each fire brigade on shift. Support groups make also at least a drill per year.

All plant personnel make a practical drill at least once each four year.

However, annual nuclear emergency drills often include scenarios complemented with fire accidents which must be solved by plant personnel and the emergency organization.

At least each four years there is a combined drill with external firemen to test the effectiveness of this coordination.

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CERNAVODA NUCLEAR POWER PLANT: MODIFICATIONS IN THE FIRE PROTECTION MEASURES OF THE CANDU 6 STANDARD DESIGN

XA9847533

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Abstract

Having as purpose the improvement of fire safety at the Cernavoda NPP - both in the prevention and the protection aspects in the case of fire - we implemented some modifications in the CANDU 6 standard design. These improvements are inspired, mainly, from two sources:

- the world-wide achievements in the field of fire protection techniques, introduced in nuclear power plants since the middle of 70's, when the CANDU 6 design was completed;
- the national practice and experience in fire protection, usually applied in industrial objectives (conventional power plants, in particular).

The absence of any incident may be considered as a proof of the efficiency of the implemented fire preventing and protection measures.

1. INTRODUCTION

In 1979, Romania, an Eastern European developing country, has embarked on a very ambitious nuclear power project with the aim to build at Cernavoda, on the Danube bank, the first nuclear plant, supposed to have, finally, 5 units of CANDU 6 type.

Romania's choice of the CANDU was based on this technology's outstanding international record for safety, environmental protection and reactor reliability. Until 1989, the normal development of the project suffered from the disadvantages of the highly centralised economy, from the decisions taken by the political leaders, sometimes, against technical or economical reasons.These aspects caused large delays in completion of the project.

After the Romanian revolution of 1989, the circumstances affecting the completion of the Cernavoda project have changed. A new development of the project has begun in December of 1990, when a Consortium made up of the Atomic Energy of Canada Limited and the Italian Company ANSALDO working closely with RENEL has established an effective new project management team.

The unit 1 construction started based on AECL standard design. Since the design had been prepared, some major incidents occurred: the incident of Browns Ferry NPP, in 1975, Three Miles Island, in 1979, Vandellos 1, in 1989 and Chernobyl accident in 1986. Fire protection has received serious attention all over the world.

2. RELEVANT SUBJECTS OF THE DESIGN UPGRADING

Having as purpose the improvement of fire safety at the Cernavoda NPP - both in the prevention and protection aspects in case of fire - we implemented some modifications in the CANDU 6 standard design. These improvements are inspired, mainly, from 2 sources:

A. The world-wide achievements in the field of fire protection techniques, introduced in nuclear power plants since the middle of 70's, when the CANDU 6 design was completed:

- The development and use of Fire Risk Analysis for the assessment of the protection measures;
- The use of modern addressable fire detection system;
- The use in civil works of new materials, which can be easily installed and provide highly efficient fire protection.

B. The national practice and Romanian expertise in fire protection, usually applied to industrial objectives (conventional power plants, in particular). The traditional approach in the fire protection design, used in the construction of industrial objectives in Romania, induced design changes at the Cernavoda NPP, too. This approach has its roots in the lack of confidence in the action of the active measures (due to the low reliability of the detection and fire suppression systems previously available in Romania), as well as in the relatively low level of the "safety culture" of the operational and maintenance personnel whose preventive attitude and skills for fire extinction were not enough developed.

The Romanian program for upgrading fire protection became substantial due to the mission organized by IAEA, at the request of the Romanian Commission for the Nuclear Activities Control, our national regulatory board.

The IAEA experts pointed out the importance of fire preventing for nuclear safety of a nuclear power plant. In fact, the fire protection system was considered as a safety system.

Based on the recommendation of the experts, as well as on the provisions of the IAEA Safety Guide 50 - D2 and the Romanian rules for fire preventing and prevention, we reviewed the CANDU 6 standard design.

Our concept is intended to implement the phylosophy of "defence in depth" protection against the hazards of a fire and its associated effects on safety related equipment, with its three main objectives:

- preventing fire from starting;
- quickly detecting and extinguishing the fire which does start, thus limiting the damage;
- preventing the spread of the fire which has not been extinguished, thus minimizing its effect on the essential plant functions.

The design of the Cernavoda NPP recognizes fire as a design basis event. The fire protection design is based on one random fire at a time, which may occur at any location in the plant.

More conservative than the Canadian philosophy which states that combustible materials are subject to ignition only whenever a source of ignition is present, our philosophy is base on the assumption that an ignition may occur even when an important thermal load only is present. Therefore, we proceeded to find adequate fire preventing and protection measures wherever the thermal load exceeds 200 MJ/mp. Thus, the first step in preparing fire preventing program was to determine thermal load for each area or room in the nuclear power plant. This was done in a Fire Risk Analysis. The Fire Risk Analysis performed before the initial loading of the reactor fuel consists of:

- identifying the areas (zone or room) with high thermal load
- analyzing of the consequences of fire in such areas, taking into account the presence of the items important to safety
- determining the required fire resistance of fire barriers
- identifying cases where aditional fire separation is required
- determining of the type of fire detection and protection means to be provided.

Based on this analysis, we establish the following aditional measures:

2.1. Fire Protection of Structural Steel in the Nuclear Service Building

The stipulations of the Romanian technical, related to fire resistance require the rating of compartment boundary to be the same with the resistance rating of the structural steel assemblies upholded. The AECL design did not observed this requirements. The structural steel being unprotected, under fire they would resist only 15 minutes, although uphold walls have a fire resistance rating of an hour for the top level and two hours for the current levels. Therefore it became necessary to use a fire protection coating on the structural steel in order to upgrade the level of fire resistance. Depending on the stage of works on site, defferent protection solutions were used: asbestos spray, glass fiber reinforced gypsum plaster, intumescent paint.

2.2. Upgrading of Fire Resistance of Fire Barriers

Some tests were carried out in order to verify that the openings in each fire barrier are protected by elements and material designed and tested to provide an appropriate fire resistance rating apropriate (fire rated doors, fire rated dampers, fire rated sealings around all the electrical and mechanical penetrations).

The fire doors as well as the fire dampers did not meet all criteria required by the Romanian norms. Since the standard design requires that a fire resistant door or damper would guarantee tightness and mechanical stability in case of fire, our approach, more conservative, asks, in addition, to provide a high thermal insulation of such doors or dampers, in case of fire. Therefore, we proceeded to substitute the fire dampers on the ventilation ducts at the penetration of the fire compartment boundary by fire resistant dampers being adequately qualified. The doors which did not assure the adequate thermical insulation were verified one by one. Some of them, for the rooms containing an important inventory of combustible materials, were coated by intumescent paint.

2.3. New Fire Compartments

Based on the Fire Risk Analysis, we defined some new fire comparments in the Service Building, smaller than those considered in the standard design, round the rooms with a large thermal load: new fuel storage area, cable spreading room. A carefully analysis was performed for these areas, as well as for the area with ventilation system filters. Also, we improved the physical separation between some redundant components, as electrical panels, Diesel generator, by erecting separation fire resistant walls.

2.4. The Upgrading of the Detection and Extinguishing Systems

A hydraulic calculation for the water piping in the spreading cables room proved that the extinguishing system did not assure an efficient fire suppressing. Therefore, the system was re-designed. Some cables in this room are fire retardant, but this does not apply to all of them. The room is protected by a deluge system which has its control valves located in the corridor in the proximity of the room. The water system is sectorized in seven inlets covering all the area of cables in the room. A modern and performant system of detection CERBERUS type was provided. As a passive protection we provided protection of the structural steel of this room. An additional measure, which is applied to all Romanian power plants, was established for the spreading room, but because of the Canadian partner's reticence it was not yet applied. This refers to fire stops from incombustible materials installed on cable trays, at a distance of 25 meters.

The fuel route in the station begins in the new fuel storage area, which is adjacent to the main air lock of the reactor building. New fuel arrives at Cernavoda packed in expanded polystyrene containers in wooden cases. The standard design provided no protection measure for the fresh fuel room. Based on the Fire Risk Analysis, some improvements were done: the structural steel have been protected by asbestos spray and the access doors have been protected by intumescent paint, a smoke detection system and a water spray extinguishing installation were provided. The personnel access to the room is carefully controlled.

In the areas containing air filters of the ventilation systems, CO2 extinguishind system was substituted by a water spray extinguishing system of the sprinkler type. Later on, taking into account the disadvantages of CO2, all over the plant, this system was replaced by the water spray system

Above the false ceilings, where a thermal load from cables was present, we provided smoke detectors.

2.5. The Upgrading Measures in the Balance of the Plant

In the turbine building, the risk rests with the lubrication oil, generator seal oil and the hydrogen cooling systems, particularly with the piping. Now, the turbine and generator bearings are protected by a manually operated water system of the deluge type. The system is actuated from two different sides. The spurious actuating is prevented. In the original ANSALDO design, the automatic operation of the same system type was specified. This current arrangements do meet the local Romanian technical standards.

Smoke detectors are installed over each bearing. Operating and training procedures are in place to minimize the delay between fire detection and the application of the suppression system.

In the standard design, turbine clean / dirty lube oil storage tank was located in a room under the main lube oil tank, without any protection against fire. Based on our norms, this tank of 92 tones capacity is separated from all other areas of the turbine building by three hour rated enclosure. A water spinkler system is installed. An oil containment system is provided as part of the protection scheme. The system is capable of containing all the oil in the tank plus the sprynkler system discharge. The oil tank is also protected also by a dedicated foam system. Some smoke detectors are provided. To remove any source of ignition, the electrical equipments are capsulated.

2.6. Fire Fighting Capabilities

Human fire fighting action represents an important part of the defence in depth strategy. For some area of the plant, human fire fighting action is the second line of defence (the first line being the automatic extinguishing system), while for other areas with lower fires load, human fire fighting represents the only method of fire extinguishing.

Based on a fire fighting approach different to that of the Canadian nuclear stations, the Cernavoda NPP has in its organizational chart a dedicated civil fire brigade composed of 44 full-time professional fire fighters. They operate in five shifts, covering 24 hour/day. This fire brigade is reporting to the Health Physics Department and is responsible for the performance of such activities as: supervision and control for fire prevention, training of the staff for fire protection maintaining the readiness of the fire fighting equipment and supplies, liaison with external military fire brigade.

Although the dedicated fire brigade exist, the operating personnel is familiar with the fire extinguishing procedures for all possible hazards, being regularly trained and examined on their knowledge of the relevant procedures.
The operational personnel is, also, trained for the evacuation in case of fire.
In the proximity of the nuclear power plant a military fire brigade operates, responsible for fire intervention and rescue in the town, villages and industrial objects of the area.
Combined drills are conducted regularly so that the off-site brigade and plant fire brigade can learn to act as a single team.

CONCLUSIONS

A comparison of the requirements related to fire protection reveals an evident flexibility which exist in Canada in the determination of the fire protection measures. The corresponding Romanian requirements appear more rigid, which proves a very conservative approach. Our main direction was the enhancement of the passive methods for fire protection. This approach has its roots in the lack of confidence in the action of the active measures, as well as in the relatively low level of the "safety culture" of the operational and maintenance personnel, whose preventive attitude was not enough developed. Despite of procuring a modern, reliable equipment for fire detection and suppression and providing an adequate training for operational personnel, the Romanian authorities and designers still emphasize the use of passive methods.
Our experience in the operation of a nuclear power plant is very limited (about one year), but the absence of any incident may be considered as a proof of the efficiency of the implemented fire preventing and protection measures.

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XA9847534

INTEGRATED APPROACH TO FIRE SAFETY AT THE KRŠKO NUCLEAR POWER PLANT — FIRE PROTECTION ACTION PLAN

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Abstract

Nuclear Power Plant Krško (NPP Krško) is a Westinghouse design, single-unit, 1882 Megawatt thermal (MWt), two-loop, pressurized water nuclear power plant. Construction of the plant was started in the mid 70's and initial criticality occurred in September 1981. NPP Krško is located on the north bank of the Sava River about 2 km southeast of Krško, Slovenia. The fire protection program at NPP Krško has been reviewed and reports issued recommending changes and modifications to the program, plant systems and structures. Three reports were issued, the NPP Krško Fire Hazard Analysis (Safe Shutdown Separation Analysis Report), the ICISA Analysis of Core Damage Frequency Due to Fire at the NPP Krško and IPEEE (Individual Plant External Event Examination) related to fire risk. The Fire Hazard Analysis Report utilizes a compliance — based deterministic approach to identification of fire area hazards. This report focuses on strict compliance from the perspective of US Nuclear Regulatory Commission (USNRC), standards, guidelines and acceptance criteria and does not consider variations to comply with the intent of the regulations. This review was constructed in accordance with the guidance set forth in Branch Technical Position CMEB 9.5-1, Appendix A. The probabilistic analysis method used in the ICISA and IPEEE report utilizes a risk based and intent based approach in determining critical at-risk fire areas. This method comprised of: Identification of potentially important fire areas; screening of fire areas based on probable fire-induced initiating events; each fire area remaining is numerically evaluated and culled on frequency; quantification of dominant areas. After the identification of these high risk areas, vital equipment affected by a fire in each area was assessed and modifications were suggested to reduce cdf (core damage frequency) for each area. Based on all above reports an extensive Fire Protection Action Plan was prepared utilizing the methodology for prioritization of proposed modifications as follows: CATEGORY 1 - CDF > 1.0E-6 event/rx-year and the potential modification(s) meets the cost benefit ratio criteria of <US$1000/person-rem to implement the modification(s). CATEGORY 2 - CDF > 1.0E-6 events/rx-year and the potential modification(s) exceeds the cost benefit ratio criteria of >US $1000/person-rem to implement the modification(s). CATEGORY 3 - CDF < 1.0E-6 events/rx-year.

NPP Krško has already completed the following suggestions/recommendations from the above and OSART reports in order to comply with Appendix R: Installation of smoke detectors in the Control Room; Installation of Emergency Lighting in some plant areas and of Remote Shutdown panels; Extension of Sound Power Communication System; Installation of a Fire Annunciator Panel at the On-site Fire Brigade Station; Installation of Smoke Detection System in the (a) Main Control Room Panels, (b) Essential Service Water Building, (c) Component Cooling Building pump area, chiller area and HVAC area, (d) Auxiliary Building Safety pump rooms, (e) Fuel Handling room, (f)

Intermediate Building AFW area and compressor room, and (g) Radwaste building; inclusion of Auxiliary operators in the Fire Brigade; training of Fire Brigade Members in Plant Operations (9 week course); Development of Fire Door Inspection and replacement program; sealing of Fire Barriers between areas; Development of Fire Response Procedures for improved response to fire events in critical areas of the plant. The above modifications, in particular the installation of smoke detectors in the Control Room, have substantially reduced the overcall plant fire-induced cdf. The most important remaining modifications to the plant include installation of a sprinkler system and fire wrapping of cables in some plant areas which will reduce the plant fire-induced cdf from 1.0E-4 events/rx-year to approximately 1.2E-5 events/rx-year which is equivalent to cdf values at US plants after implementation of Appendix R criteria.

1.0 Introduction

Nuclear Power Plant (NPP) Krško is a Westinghouse- designed, single-unit, 1882 megawatts thermal (MWt), two-loop, pressurized water reactor (PWR). Construction of the plant was started in the mid-1970s and initial criticality occurred in September 1981. NPP Krško is located on the north bank of the River Sava about two kilometers (km) southeast of Krško, Slovenia.

The purpose of the Fire Protection Action Plan (FPAP) is to prioritize proposed fire protection modifications contained in the NPP Krško Fire Hazards Analysis - Safe Shutdown Separation Analysis (SSSA) (Ref. 1), the International Commission for an Independent Safety Assessment (ICISA) Analysis of Core Damage Frequency Due to Fire at the Krško Nuclear Power Plant (Ref. 2), and the Operational Safety Review Team (OSART) (Ref. 3) reports using a risk-based approach which will provide for a timely reduction of the probabilistically-significant contributors to fire-induced core damage frequency (CDF). A cost benefit analysis has been performed for proposed modifications in fire areas, with the exception of the main control room, which were found to have fire-induced CDF exceeding 1.0E-6/ry (Ref. 4). The action plan utilizes event sequences and system models from the Krško IPE report (Ref. 5) and the resultant fire risk from the Fire IPEEE Level 1 and 2 reports (Refs. 4,6).

The SSSA report used a compliance-based, deterministic approach to identification of fire area hazards. The report focused on strict compliance from the perspective of United States Nuclear Regulatory Commission (USNRC) standards, guidelines, and acceptance criteria and did not consider variations to comply with the intent of the regulations. The review was conducted in accordance with the guidance set forth in Appendix A of "Guidelines for Fire Protection for Nuclear Power Plants" (Ref. 7). The purpose of the study was to perform and document a comprehensive analysis of the separation between redundant safe-shutdown components and cables in the context of post-fire shutdown system separation requirements defined by Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, 10CFR50, Appendix R (Ref. 8) and ancillary USNRC regulatory guidance.

The ICISA report used a probabilistic screening analysis method in determining critical at-risk fire areas. This method was comprised of the following steps:

(1) *Identification of potentially important fire areas:* Fire areas which have either safety-related equipment or cables were identified as requiring further analysis. Critical safety components required for hot standby and cold shutdown within these fire areas were identified. Areas not containing vital equipment were screened from further analysis.

(2) *Screening of fire areas based on probable fire-induced initiating events:* Estimation of the fire frequency for all critical plant locations and identification of the resulting fire-induced initiating events and "off normal" plant states were performed. Those areas with a random failure probability of less than 1.0E-3 were screened from further analysis.

(3) *Each fire area remaining was numerically evaluated and culled on frequency:* Fire area specific initiating event frequencies were used to screen the remaining areas with a low frequency of initiating events. The only fire areas remaining had contributions to CDF of greater than 1.0E-6/ry.

(4) *Quantification:* After the screening analysis eliminated all but the probabilistically significant fire areas, quantification of dominant areas was completed as follows:

(a) The temperature response in each fire area was estimated for each postulated fire. The fire growth code COMPBRN (Ref. 9) with some modifications was used to calculate fire propagation and equipment damage.

(b) A recovery analysis was performed which accounted for recovery of non-fire related random failures.

(c) The probability of barrier failure for adjacent critical fire areas was evaluated.

The OSART review consisted of an international team of experts which performed a review of plant practices, including a review of the fire protection program. Detailed in their report were recommendations concerning improvements to the fire protection program associated with the
fire brigade, fire emergency procedures, fire barrier penetrations, training, fire cabinets, and fire brigade notification for response.
It has been found that implementation of Appendix R separation guidelines led to approximately one order of magnitude reduction in fire-induced CDF at United States light water reactors (Ref. 10). While Appendix R implementation has been found to substantially reduce fire risk on a plant-wide basis, most plant areas at NPP Krško (Ref. 4) have sufficient redundant methods of safe shutdown (i.e., critical safety-related equipment located in other plant areas). Therefore, Appendix R compliance is required in limited plant areas if the only goal is to accomplish a low fire risk.

2.0 Fire Protection Modifications Completed to Date

Many modifications have already been implemented at NPP Krško based on the recommendations contained in the SSSA, ICISA, and OSART reports. These modifications, in particular the installation of in-cabinet smoke detectors in the control room, have already led to a substantial reduction in risk (Ref. 4).

1. Installation of in-cabinet smoke detectors in the control room
2. Installation of emergency lighting at the evacuation panels and other plant areas

3. Inclusion of auxiliary operators in the fire brigade
4. Training of fire brigade members in plant operations (nine-week course)
5. Installation of a fire annunciation panel at the on-site fire brigade station
6. Development of a fire door inspection and test program
7. Installation of additional smoke detectors in many plant areas
8. Sealing of fire barriers between areas
9. Development of a modification package to upgrade and correct fire door deficiencies
10. Development of fire response procedures for those areas which were found to be the most risk-significant (Ref. 4)

3.0 Prioritization Method

The prioritization method for the remaining fire-related plant modifications is based upon providing for a timely reduction of the overall fire-induced CDF. This method ranks proposed modifications for a fire area into the following three categories:

1. Category 1 - CDF > 1.0E-6/ry and the proposed modification(s) meets the cost benefit ratio criteria of < U.S. $100,000/person-Sievert reduction (U.S. $1,000/person-rem reduction) to implement

2. Category 2 - CDF > 1.0E-6/ry and the proposed modification(s) exceeds the cost benefit ratio criteria of < U.S. $100,000/person-Sievert reduction (U.S. $1,000/person-rem reduction) to implement

3. Category 3 - CDF < 1.0E-6/ry

A CDF contribution of 1.0E-6/ry from a functional accident sequence (a combination of an initiating event together with functional failures resulting in core damage) meets USNRC reporting criteria for IPEEE analyses (Refs. 11,12). Functional accident sequences with frequencies less than 1.0E-6/ry are deemed to be insignificant contributors to risk. The cost-benefit ratio of U.S. $100,000/person-Sievert (U.S. $1,000/person-rem) is the USNRC regulatory value used in back-fit rule calculations and also employed for evaluating generic and unresolved safety issues.

For Category 1, the modifications are highly recommended to be implemented as soon as funding and resources are available. For Category 2, the modifications should be scheduled following completion of Category 1 modifications. For Category 3, the modifications are not significant contributors to fire-induced CDF and should be scheduled to comply with Appendix R criteria at plant management's discretion.

The cost analyses used are comprehensive and follow the guidelines of NUREG/CR-3568, "A Handbook for Value-Impact Assessment," (Ref. 13) and NUREG/CR-4627, Revision 1, "Generic Cost Estimates," (Ref. 14). The computer code FORECAST 3.0 (Ref. 15), which incorporates this knowledge-based information, was used to develop cost estimates for the proposed Category 2 plant modifications (with the exception of the main control room).

Cost analyses for the various tasks required for each proposed plant modification are performed according to standard engineering practices. This involves an initial design evaluation of the plant modification, identification of equipment and materials necessary for the modification, and an assessment of the work areas within the plant in which the proposed modification will take place. All plant cost estimates are presented in 1995 US dollars and represent implementation costs for the specific improvements, (i.e., one-time cost incurred by the nuclear power plant). There are no annual costs, (i.e., recurring costs), associated with any of the proposed modifications.

In addition to the cost of physical modifications, the cost analyses include costs for engineering and quality assurance, radiation exposure, health physics (HP) support, and radioactive waste disposal. Nuclear power plant costs associated with re-writing operating and testing procedures, staff training, and other technical tasks are also considered.

4.0 Core Damage Frequency (Level 1) Methodology and Results

The CDF quantification is based on the Individual Plant Examination of External Events (IPEEE) Fire Probabilistic Safety Assessment (PSA) which was completed in June 1996 (Ref. 4). The overall methodology used in the development of the Krško Fire IPEEE conforms with the guidance provided by USNRC Generic Letter (GL) 88-20, Supplement 4 (Ref. 11) and the detailed guidance provided in NUREG-1407 (Ref. 12). The fire PSA methodology followed the same approach used in the NUREG-1150 fire PSAs for the Surry and Peach Bottom plants (Ref. 16) and subsequent studies of the LaSalle and Grand Gulf plants (Refs. 17,18).

The methodology made use of past PSA experience (Refs. 19,20), generic databases, and other defensible simplifications to the maximum extent possible. The Krško Fire IPEEE is consistent with the Krško Individual Plant Examination (IPE) internal events analysis (Ref. 5) in that the same event trees, system success criteria, and recovery analysis assumptions were used.

The general methodology consisted of an initial plant visit, screening of fire areas to identify locations having the potential to produce risk-dominant fire sequences, and quantification. The initial plant visit is used to determine the general location of cables and components for the systems of interest, verify the physical arrangement of fire areas, and to complete fire area checklists which aid in the screening and quantification steps. The initial plant visit also includes confirmation from plant personnel that current documentation is being utilized, as well as clarification of questions which may have arisen during the walkdown. Finally, as part of the initial plant visit, a thorough review of fire-fighting procedures is conducted, including consideration of manual fire suppression by the fire brigade.

The screening analysis includes three sub-tasks. First, potentially important fire areas are identified. These are areas that have either safety-related equipment or power and control cables for that equipment. Fire areas are designated as portions of buildings that are separated from other areas by boundaries acting as rated fire barriers, except for certain outdoor areas which are provided with spatial separation from other fire areas. Fire barriers were defined based on the "as-built" condition of the plant. A large number of fire areas were screened out by inspection based on the absence of safety-related equipment or power or control cables for such equipment in the fire area.

The second step is to screen areas where fires would only lead to a fire-induced initiating event at a lower frequency than the corresponding internal events cause (Ref. 5). The third and final step in screening is to numerically evaluate the remaining fire areas and cull on frequency so that only fire areas which are potentially capable of yielding fire-initiated

TABLE 1. CORE DAMAGE FREQUENCY RESULTS, NPP KRŠKO (BEFORE AND AFTER FPAP IMPLEMENTATION)

Fire Area	Fire Scenario Description	Base Case CDF (Per Reactor-Year, Before FPAP)	CDF After FPAP (Per Reactor-Year)
CB-1	Fire-induced abandonment of the MCR (due to smoke obscuration), loss of feedwater, and either failure of recovery from the evacuation panels or a fire in the main benchboard from which recovery is not possible	8.7E-5	1.2E-5
AB-9	AB (El. 94.2 m) fire, fire-induced failure of component cooling system and the positive displacement charging pump, and random failure of either RCS cooldown or the turbine-driven AF system and the feedwater system	7.7E-6	<1.0E-7
CB-3A	Emergency switchgear room A fire, fire-induced loss of feedwater, fire-induced loss of both motor-driven AF pumps, random failure of the turbine-driven AF pump and feed- and-bleed cooling	4.1E-6	<1.0E-7
SW	Essential service water building fire resulting in fire damage to pumps A & B, and random failure of either the turbine-driven AF pump and the feedwater system, or RCS cooldown and the positive displacement charging pump	3.2E-6	<1.0E-7
AB-3	AB (El 100.3 m) fire, fire-induced multiple spurious actuations of motor-operated valves powered from MCC 221 leading to an RCP seal LOCA and loss of instrument air; train A scenario (loss of instrument air) frequency is 2.1 E-7, train B scenario (RCP seal LOCA) frequency is 7.2 E-7	9.3E-7	9.3E-7
CC	Component cooling system building fire resulting in fire damage to pumps A & B, and random failure of either the turbine-driven AF pump and the feedwater system, or RCS cooldown and the positive displacement charging pump	8.4E-7	<1.0E-7

sequences with CDF contributions of greater than 1.0E-7/ry remain. A number of discrete initiating events were considered as appropriate, including loss of essential service water, loss of component cooling water, loss of instrument air, transients with and without main feed water, loss of a direct current (DC) bus, loss of offsite power, small loss of coolant accident (LOCA), and medium or interfacing system LOCAs due to spurious actuation of relief or isolation valves.

The quantification step involved detailed analyses of the potentially dominant fire-initiated accident sequences identified during the screening analysis. Quantification considered the temperature response in each fire area for each postulated fire, recovery analysis, and fire barrier failure analysis. Temperature response was modeled using the latest version of the COMPBRN fire growth code (Ref. 9). Recovery analysis considered, consistent with the internal events IPE, recovery of non-fire-related random failures. The barrier failure analysis considered potential combinations of adjacent fire areas which could result in core damage sequences.

Throughout the fire analysis, fire-related generic issues were addressed. Such issues were raised in the "Fire Risk Scoping Study" (Ref. 10) and in a report addressing Generic Issue 57 (GI-57), "Effects of Fire Protection System Actuation on Safety-Related Equipment" (Ref. 21). These issues include control systems interactions, total environment equipment survival, manual fire brigade effectiveness, inadvertent and advertent fire protection systems (FPSs) actuation, and seismic/fire interactions.

Six fire areas were found to have core damage frequency contributions of greater than 1.0E-7/ry before implementation of modifications. After implementation of FPAP modifications, only two such areas are identified. The pre- and post-implementation CDF results for NPP Krško are shown in Table 1.

5.0 Level 2 Methods and Results

The fire analysis containment event tree (CET) quantification methodology (Ref. 6) was consistent with the internal events IPE containment analysis (Ref. 5). The same bridge trees, containment system success criteria, CETs, and recovery analysis assumptions were used in the fire Level 2 analysis as in the internal events IPE.

The structure of the Level 1 fire IPEEE did not address containment heat removal (CHR) systems in the CDF quantification. Thus, it was necessary to construct a containment system tree (CST) or bridge tree to complete the system probabilistic analysis and to define the plant damage states (PDS) needed to perform source term analyses. The steps used to quantify the PDS are:

1. Quantify the CDF and obtain the dominant contributors
2. Construct the CST
3. Determine the success criteria for the top events in the CST
4. Define the PDS
5. Assign (and "bin" together) the dominant core damage sequences into the PDS
6. Link the containment systems fault trees to the dominant core melt sequences and quantify the frequency of each PDS

The PDS is a function of specific plant characteristics important to containment performance. These plant characteristics include the following:

1. Level 1 fire-induced initiating event (small loss of coolant, S, or transient, T)
2. Time of core melt (less than or greater than four hours; early, E, or late, L)
3. Core melt and reactor pressure vessel (RPV) failure pressure (high, H, or low, L)
4. Status of the emergency core cooling system (ECCS) (injection before reactor vessel failure, B; injection after reactor vessel failure, A; no injection, N)
5. Status of containment heat removal, addressing containment spray injection (CSI), containment spray recirculation (CSR), and RCFC (containment heat removal success, Y; containment heat removal failure, N)
6. Status of containment (initially intact, N; containment isolation failed, I)

The CET developed in Reference 22 was used for the fire scenarios. In addition, the source terms corresponding to each release category were maintained. Each PDS is processed through the CET. At each CET node a probability was developed which describes the confidence of the analysts that the event will or will not occur for the accident sequence under consideration. Once the nodal probabilities are developed for a PDS, the CET is quantified by multiplying the component probabilities and determining the probability of occurrence of

TABLE 2. NPP KRŠKO FIRE PSA LEVEL 2 RESULTS

Release Category Number	Release Category Definition	Base Case Release Frequency (/ry)	Release Frequency After FPAP (/ry)
—	Core Damage Frequency	1.0E-4	1.2E-5
1	Core Recovered In-Vessel; No Containment Failure	0	0
2	No Containment Failure	2.1E-6	3.2E-8
3A	Late (Beyond 24 Hours) Containment Failure, No Molten Core-Concrete Attack	3.5E-7	4.0E-8
3B	Late (Beyond 24 Hours) Containment Failure, Molten Core-Concrete Attack	8.1E-5	9.7E-6
4	Basemat Penetration (No Overpressure Failure)	6.8E-7	9.3E-8
5A	Intermediate (From End of Rapid Debris-Coolant Interaction Ex-Vessel to 24 Hours) Containment Failure, No Molten Core-Concrete Attack	8.5E-6	9.7E-7
5B	Intermediate From End of Rapid Debris-Coolant Interaction Ex-Vessel to 24 Hours)Containment Failure, Molten Core-Concrete Attack	1.2E-7	1.4E-8
6	Early (From Onset of Significant Zirconium Oxidation Reaction Until Vessel Failure and End of Containment Dynamic Response to Vessel Failure or End of Rapid Fuel Debris/Coolant Interaction Ex-Vessel) Containment Failure	1.7E-7	1.9E-8
7A	Isolation Failure, No Molten Core-Concrete Attack	1.2E-6	1.2E-7
7B	Isolation Failure, Molten Core-Concrete Attack	9.3E-6	1.2E-6
8A	Bypass, Scrubbed	0	0
8B	Bypass, Unscrubbed	0	0

TABLE 3. NE KRŠKO FIRE PROTECTION ACTION PLAN COST-BENEFIT ANALYSIS RESULTS

Fire Area	CDF (per reactor-year)	Suggested Modifications	Person-Rem Release Before FPAP	Estimated Cost	CDF After FPAP (per reactor-year)	Person-Rem Release After FPAP
CB-1	8.7E-5	Double fusing of control circuits for separation and add circuits to shutdown panels (minimal list)	42.0	by NEK	1.2E-5	6.0
AB-9	7.7E-6	Install area sprinkler system and one-hour wrap (nine critical cables)	3.8	$359,000	<1.0E-7	0.0
CB-3A	4.1E-6	Install three-hour cable wraps (nine cables)	2.9	$63,000	<1.0E-7	0.0
SW	3.7E-6	Install sprinkler system over SW motors, install one-hour heat shield between pumps and three-hour cable wraps (three power cables)	1.5	$101,000	<1.0E-7	0.0

each release category conditional on the occurrence of the given PDS. The process is repeated for all PDS.

The Krško fire PSA release categories are defined in Table 2. The table also shows the results of the PDS mapping by providing the frequency of the release category. Frequencies are shown for the "base case" (as-found condition of the plant) as well as for the case which will exist after implementation of the FPAP.

After implementation of the FPAP recommendations the release category frequencies are being dominated by the control room abandonment fire scenario. The only other remaining unscreened fire area AB-3 contributes less than eight percent to the total.

Four containment release categories were found to contribute 97.5% to the total CET frequency before modifications. After implementation of the modifications, three release categories contribute the same percentage to the reduced frequency total.

6.0 Level 3 Methods and Results

The MAAP4-DOSE code was used to calculate the consequences of the release categories, using site population and meteorological data. The resulting consequences, in person-rem per reactor-year, were multiplied by twenty to account for 20 years of remaining plant life. The risk results were calculated before and after implementation of modifications. The 20-year risk from fire-induced severe accidents before the modifications was 49 person-rem (0.49 person-Sieverts). After implementation of the modifications, the 20-year risk was 6.1 person-rem (0.061 person-Sieverts), a reduction by a factor of more than eight. The Level 3 and cost-benefit results are shown in Table 3.

7.0 Follow-Up Actions

Based on the recommendations in the Fire Protection Action Plan, engineering and design changes have been initiated and are in progress for the four Category 2 fire areas. The modification packages include design input/bases, safety evaluations, required FSAR changes, supporting analyses and calculations, design information and drawings, changes to plant procedures, and installation and test procedures. The general type of modifications involve circuit isolations, cable fire-wrapping or rerouting, and/or addition of sprinkler systems.

8.0 Conclusions

Using a risk-based methodology, the Fire Protection Action Plan evaluated and prioritized fire protection program improvements suggested in three independent reviews. The goal of the prioritization was to identify those improvements which provided the greatest risk reduction and assign the highest priority to such improvements.

Some plant fire protection program improvements and plant modifications have already been completed. These modifications, in particular the installation of smoke detectors in control room cabinets, have already substantially decreased the fire-induced CDF.

Based on the FPAP review and prioritization, the most important remaining modifications to the plant to reduce overall fire-induced risk are the Category 2 modifications for fire areas

CB-1, CB-3A, AB-9, and SW. These modifications are: (a) circuit isolation of vital equipment located on the evacuation panels and MCCs and the addition of RCS wide-range temperature and PORV control at the evacuation panels; (b) providing a three-hour fire wrap for train B cabling in fire area CB-3A; (c) installation of a sprinkler system and fire wrapping of cables in the AB basement; and (d) installation of a sprinkler system above and heat shield between the SW pumps, and fire wrapping of SW pump power cables. Implementation of these modifications will reduce the fire-induced CDF from approximately 1.0E-4 /ry to approximately 1.2E-5 /ry, which is equivalent to fire-induced CDF values at western nuclear power plants after implementation of Appendix R criteria. Design change packages to implement the Category 2 modifications are currently being developed.

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FIRE PROTECTION PROGRAMME DURING CONSTRUCTION OF THE CHASHMA NUCLEAR POWER PLANT

XA9847535

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Abstract

A clear view is given of several measures that have been taken with regard to fire prevention, protection and fire fighting during all phases of the construction, installation and commissioning of the Chashma nuclear power plant to protect personnel and equipment so that any delays in plant operation as a result of fire incident can be avoided. These measures include the precautions taken, the provisions made for fire exstinguishers and hydrants, and the setting up of a fire brigade. An overview is also given of the fire incidents that have occurred.

While under construction, a Nuclear Power Plant does not differ in any fundamental respect from, for example a conventional steam power plant or any other industrial plant of similar size as regard fire protection systems. Rather a high density of valuable equipment and components is reached during the installation phase. Any delay in start of commercial operation, justify an above average level of fire protection on the site also during the erection of the plant.

This paper outline fire protection programme at Chashma Nuclear Power Project site, beginning from start of construction activities through erection and installation.

This include fire protection management and organization, precautions during construction and erection, provisional fire fighting system, establishing fire brigade, off site assistance in fire fighting, over view of fire incidences occurred at site etc.

Fire Protection Management and Organization

At the very outset of the plant construction (early 1992) for example with mobilization of the main contractor, a fire protection organization was formed at the site, with the Deputy General Manager (Construction) as overall responsible for the fire prevention programme assisted by a "Fire Protection Manager", a "Fire Officer" and a "Fire Squad". In view of the type of contract for the plant, the responsibility of implementation of fire protection at main plant buildings, preparation areas, workshops, residential areas of expatriates and local contractors, is that of the main contractor.

Organization of Main Contractor

Following the international and national guide lines for the fire protection of nuclear power plants, fire protection and fire fighting organization was also formed by the main contractor with the collaboration of sub-contractors. The same is lead / headed by a qualified and experienced fire manager assisted by the fire officers of sub-contractors.

The Fire Protection Manager is a Sr. Officer with sufficient fire protection engineering experience and knowledge of radiological protection.

Similarly Fire Officer have sufficient fire prevention, protection and fire fighting experience and knowledge. While fire squad comprised of trained workers.

The organization has been sub-divided into three level i.e. company, section and team. Each level is responsible for their own work area. At the company level each sub-contractor has appointed a fire officer and each section in the company have fire fighters responsible for fire prevention and fire fighting in their respective job areas.

Each work team in a section has at least one fire fighter responsible for fire prevention at their specific work area.

Responsibilities of the Owner

Chashma Nuclear Power Plant management is responsible for overall management and implementation of the progrramme, through audit and inspection of contracto's fire protection / prevention plan and activities.

Joint inspection / audit visits of the site is carried out to ensure the implementation of the programme.

The implementation of the programme in office buildings used by CNPP project management and construction staff is sole responsibility of CNPP.

Owner Fire Brigade Facilities

Chashma Nuclear Power Plant fire brigade with full facilities i.e. two nos. of FIRE TENDER, two AMBULANCE and tranined FIRE FIGHTERS and qualified COMPOUNDERS" is avaiable at site round the clock.

Emergency telephone at the fire brigade post has been well displayed to the all parties at site.

Precautions Taken

The fire protection measures are based upon, devising the preventive measures with the aim of

a) minimizing the fire loading
b) preventing the out break of fire
c) limiting the spread of fire and localizing the damage
d) arranging appropriate fire fighting devices and measures which are brought into action upon out break of fire.

Accordingly the following precautions have been taken with the advancement of construction and erection activities:

a) **Good House Keeping:** The construction site and the workshop areas are kept clean and orderly, through periodic surveillance and rounds by CNPP fire officer and his counter part of contractors organization. The combustibles used are limited. The waste materials are collected and disposed of as soon as possible. The material used in construction i.e. form work, scaffolding, decking etc. are made of non-consumable and fire retardent treated wood.

All kind of stores / warehouses housed with combustible are established out side the plant buildings at a safer distance from the plant buildings and also from each other. International codes are followed in storing goods in warehouses.

b) **Issuance of Fire Permit:** To prevent outbreak of fire special attention is being given to welding and cutting operations. The operation involving use of open flames is properly supervised. In particular, written prior permission "FIRE PERMIT" is being issued by fire manager of the contractor, and welding supervisor/firemen ensure that proper fire protection and prevention measures are taken for the work. A fireman with appropriate type of fire extinguisher is always available for all welding / cutting operations which could be dangerous. Such operations are under taken only after all combustible materials that can be removed has left the area and those which could not be removed are covered with fire resistance tarpaulin. Thorough inspections of the vicinity and potentially affected zones are made immediately after the termination of each work.

c) **Patrolling During Off-Hours:** Fire and security men frequently patrol the buildings after working hours to cope with the possibility of delayed ignition which could be reason for out break of fire.

d) **Prevention of Unauthorized Entry:** The construction site has been enclosed by barbed wire, perimeter lighting installed where ever required and the main entrances are manned to prevent prospective of intruders.

e) **Provision of Fire Extinguisher:** With the start of construction work, portable fire extinguisher of appropriate type and capacity were provided at all location. Fire groups with individuals having basic knowledge about type of fire, use of fire extinguishers and action to be initiated on out-break of fire were set up.

f) **Enhancement of Facilities:** The number and places of appliances are increased in step with the progress of work at site.
A provisional system of fire hydrants was made available well before installation of plant equipment and components in the main buildings i.e. N.I. and C.I. The system consisted of a main ground level water tank, two auxiliary ground level tanks, pressurized fire water main, fire hydrants / stand pipes and hose reels at "appropriate location on all levels of the buildings". Besides, hydrants with hose reels are also provided outside around the main plant buildings, the warehouse, auxiliary boiler house etc.

g) **Traing of Fire Crew and Plant Personnel:** With the collaboration of local Civil Defense Authority and National Institute of Fire Technology, Islmabad, class training is conducted for fire fighters, security men and operation and maintenance personnels besides class and field training conducted by senior colleagues.

One site fire drills (dry and wet) are regularly held to cope with any fire incident efficiently and effectively.

Over View of Fire Incidences Occurred at Site

Since start of construction activities about twelve insignificant fire incidents of small fires have been occurred in different places due to various reasons.

The incidents occurred in main plant buildings attributed to waste material such as jute bags, wooden scrap, packing material, welding and unauthorized use of electric heater. No loss to goods, building, equipment or human was occurred, due to the deployment of effective fire fighting facilities. The incidences occurred out side in open areas was due to shrubs and dry grass etc.

Experience gained from these fire incidents is that, more attention was given to house keeping, vigilant patrolling inside the plant buildings to notice the waste material and to remove the same on priority basis. Welding and cutting operation are carried out under strict supervisions. Unauthorized use of making fire and smoking in the construction premises is strictly banned and penalty upto US $ 500 has been fixed with the termination from job for defaulters.

Wild growth and shrubs are cleared regularly all around the plant buildings. Patrolling of the site round the clock by firemen and security personnel intensified. Nos. of fire extinguishers increased. Plant fire brigade is kept on alert round the clock.

As a result of these improvement the frequency of occurrence of fire incidents is reduced considerably.

OPTIMIZATION OF EXTINGUISHING AGENTS FOR NUCLEAR POWER PLANTS

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Abstract

Focus is placed on use of extinguishing agents in nuclear power plants. The advantages and disadvantages of these agents are compared. Further perspectives for using particular extinguishing agents in nuclear power plants are outlined.

Introduction

Fire protection is an indivisible part of power plant´s nuclear safety. A fire in the plant can cause big material damage, electrical power production failure, but it may also cause an accident and radioactive leakage to the atmosphere. This is why a maximum care for the fire protection from designing to decomissioning of a power plant is so neccessary.

In my talk, I would like to focus profoundly on the last phase of fire protection - extinguisthing.

Talking abour particular extinguishing agents I will focus on their present use at PWR NPPs, their perspectives and disadvantages of their use.

Water

It is the extinguishing agent used the most often in power plants. It is used in outer unit transformer fixed extinguishing devices in the form of water fog and in fixed extinguishing equipments protecting cable areas and Diessel-generator stations.

Water can also be used in the form of a water stop as an agent dividing fire-dangerous places where other forms of division are technically impossible or too expensive.

Water is available at all the buildings of a nuclear power plant through the water *distribution systems and through underground and overhead hydrants.*

Water can be used for voltage electrical equipment extinguishing. No devices and extinguishing equipmet have been designed for larger voltage electrical equipment fires.

Devices imponant for safety reactor shut-out and residual heat withdrawal cannot be turned off nor in the case of fire. In the EGU Bechovice, there has been an experimet carried out proving that voltage electrical equipment can be extinguished by water in case of keeping certain technical conditions and space-gaps. Based on this experiment, the Slovak Standardization Authority allowed some Slovak Technical Standards exeptions for voltage electrical equipment extinguishing in Slovak NPPs .

Perspectives

Water will be the most often used extinguishing agent at NPPs, henceforth. It is also possible to upgrade its extinguishing effect by adding modern foam creating concentrates and wetting agents (AFFF, Pyrocool).

Usage of new forms of extinguishing Micro drop, Infex.

Disadvantages

Arising of contaminated water in the primary circuit and thus rising amount of radioactive waste. Water extinguishing and using boric acid at the same time brings the danger of boric acid dilution.

Foam

Foam is used mainly for inflammable liquids extinguishing at nuclear power plants and in older projects also for cable areas fixed extinguishing equipment. Highly expansive foam from mobile foam agregates is advantageous for being used for filling cable areas up in case of fixed extinguishing equipment failure or in case of fire in oulet cable areas which are not equiped by fixed extinguishing equipment. Nowadays, foam is also proposed to be the extinguishing agent for protective fixed extinguishing equipment for Diessel-generator stations (Siemens). Foam is unreplaceable in case of outer transformers fixed extinguishing equipment failure. Foam fixed extinguishing equipment is also advantageuos for main turbo-generator oil tanks and oil feeding pump systems protection.

Physical, chemical and extinguishing features of foamers used in nuclear power plants has been analysed. It is necessary to pay higher attention to choosing foam creating concentrates for risky places focusing on their service life, effectiveness and their neutralization after the service life elapse.

Perspectives

For its great inflammable materials extinguishing qualities and since there is a lot of such materials in a nuclear power plant, mobile equipment and also fixed extinguishing equipment foam wil be mainly used in future.

Disadvantages

Rising of radioactive waste amount in the primary circuit. Necessity to replace the foamer after its guaranatee term.

Extinguishing powders

Experience has showed it is not suitable to install powder fixed extinguishing equipment in power plants. The use of powder is concentrated into extinguishers. It is also useful to have some powder supplies in a fire-extinguishing tanker mainly for imflammable materials, cable areas and voltage electrical devices extinguishing.

Perspectives

Powders are supposed to be used broadly in nuclear power engergetics in case of fast neutron reactors with a liquid sodium cooler.

Disadvantages

Invalidation of extinguished electronic devices and rising of a radioactive waste amount in the controlled zone.

Carbon dioxide

Carbon dioxide is used mainly in extinguishers for electric voltage devices. After the use of halogens had been reduced the importance of carbon dioxide rised also for fixed extinguishing equipment for example oil systems (NPP Temelín) and for main circulating pump deck extinguishing projects (Siemens).

Perspectives

Henceforth, carbon dioxide will be used mainly in extinguishers. It may be used as a possible replacement of halogens in fixed extinguishing equipment.

Disadvantages

Comparing to other alternatives, there is a higher extinguishing concentration necessary which is of fatal danger for the service personnel. Extinguishing agents contributing to strenghtening the greenhouse effect (GWP) are being reassessed nowadays.

Halogens

Halogens are extinguishing agents very suitable for nuclear power plants. Their usage is limited by the 1987 Montreal Protocol. It is a nearly unreplaceable extinguishing agent in nuclear plant conditions. They are used for voltage electrical devices, electronic devices and important cable areas extinguishing.

Model experimets with halogens and halogen alternatives should verify the effectiveness of closed room flame extinguishing and their ability to prevent reflaming of infalmmable materials. These experiments are being proposed since the information on extinguishing abilities of halogen hydrocarbons available till now are tested by Cup Burner Test. According to the search of literature, room extinguishing concentrations are designed with 20% reserve. Since from the point of view of chemical mechanism the alternative agents effects are lower than the halogen ones, the experiment should verify the extinguishing agent's ability of volume extinguishing. There is to be halogen 1031 compared with halogen alternatives (CAE-410, FM 200, Halotron II) within the experimet.

Perspectives

The use of halogens at nuclear power plants was not openly prohibited in Slovakia. Halogen 1301, used mostly in fixed extinguishing equipment, is not available at the market. Considering demanded extinguishing abilities, non rising of a radioactive waste amount and non invalidation of extinguished devices, halogens are still a very perspective extinguishing agent in Slovakia.

Disadvantages

Damage of the ozon layer (ODP), contibution to the greenhouse effect (GWP) and proportionally long term of life in the atmosphere (AL).

Conclusion

In conclusion, it can be claimed that there are all kinds of extinguishing agents used within modern fire protection in conditions of nuclear power plants. We have to undertake several aspects using extinguishing agents for manual extinguishing, fixed extinguishing equipment and fire-extinguishing tankers. The extinguishing agent has to have good extinguishing effects, must not divaluate the extinguished material, must not support further rise of contaminated waste amount and must not have an enviromental impact.

Having undertaken all these aspects we have to choose the most suitable extinguishing equipment for the particular area of the plant.

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CLOSING SESSION

Chairperson

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IAEA-SM-345/46

CURRENT AND FUTURE IAEA ACTIVITIES IN THE AREA OF FIRE SAFETY

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Abstract

The paper provides a brief overview of the IAEA programme on fire safety carried out in the period 1993-97. The main results of past activities are highlighted with appropriate references to IAEA publications and other papers presented at this Symposium. Ongoing work is presented in more details. This presentation addresses two IAEA documents on the use of fire safety related operational experience in fire safety assessment currently in preparation. The first document describes potential applications, related data requirements, current reporting practice and the use of experience based data in plant specific analyses. The second document presents the root cause analysis tailored toward its application to the investigation of fires. The paper describes the main objectives, intended scope and contents of these two documents and status of document development tasks. Some insights obtained so far are outlined. Another ongoing task addressed in the paper is the development of a list of generic technical safety issues which limit the current capabilities of fire safety assessment. The paper describes the approach proposed by the IAEA in the compilation of a list that would represent broader international views. Finally, the paper addresses the future IAEA activities in the area of fire safety. These concentrate on the completion of ongoing work on guidelines and good practice documents, exchange of technical information in the area of fire safety at nuclear facilities, and fire safety review missions and training.

1. INTRODUCTION

The IAEA initiated a comprehensive programme on fire safety at nuclear power plants in 1993. The first project conducted within the framework of this programme devoted to "Fire Safety" was successfully carried out by the end of 1996. As of the beginning of 1997 the fire safety related activities were included in two new projects, one on "Current Engineering Safety Issues" and the other on "Engineering Safety Advisory Services".

The IAEA programme is intended to provide assistance to Member States in improving fire safety in nuclear power plants. In order to achieve this general objective, the IAEA programme aims at the development of guidelines and good practices, the promotion of advanced fire safety techniques, the exchange of state of the art information between practitioners and the provision of engineering safety advisory services and training in the implementation of internationally accepted practices.

2. OVERVIEW OF THE PROGRAMME AND ITS RESULTS

2.1. Development of guidelines and good practices

During the period 1993-94, the IAEA programme concentrated on fire safety and fire protection of operating plants with the main focus on the development of guidelines and good practice documents. One of the first tasks was the development of a Safety Guide [1] that formulates specific requirements with regard to fire safety of operating nuclear power plants. Draft document prepared in 1995 is being in the review for final acceptance of Member States.

A more detailed information on this document is provided in the paper entitled "Overview of IAEA published guidelines for fire safety inspection and operation in NPPs" (No. 28) presented at this Symposium.

Several good practice documents [2-4], which provide advice on *fire safety inspection,* were developed to assist in the implementation of this Safety Guide. These documents have already been published in IAEA NUSS Series as Safety Practices. These publications address al technical aspects of fire safety inspection at NPPs including fire protection measures and fire fighting capability[2], fire protection system organization, management and procedural control [3], and evaluation of fire hazard analysis [4]. A more detailed information on this document is provided in the paper No. 28 presented at this Symposium.

In 1995-96 the programme concentrated on the development of good practices in the preparation of *fire safety analysis.* Two documents providing advice on the preparation of systematic fire safety analysis at NPPs have been developed and are being prepared for publication in Safety Reports Series: "Preparation of Fire Hazard Analyses for Nuclear Power Plants" [5] and "Treatment of Internal Fires in Probabilistic Safety Assessment of Nuclear Power Plants" [6].

The first report is intended to facilitate the preparation of a deterministic analysis of fire hazard. The second report describes in detail good practices in the preparation of probabilistic safety assessment for fires at NPPs. It is intended to facilitate the implementation of a risk based approach in fire safety assessment for NPPs based on current practical experience gained in this area. A more detailed information on these two documents is provided in the paper No. 28.

In 1997 new tasks were initiated in the area of practical application of *fire safety related operational experience* at NPPs. Two good practice documents devoted to the collection and use of operational experience are under development: one on the collection and use of fire safety related operational experience and another on root cause analysis of fire safety related events. A more detailed description of these ongoing tasks will be provided in this paper. Development of another IAEA document addressing lessons learned from fire events is planned for the period 1998/99.

2.2. Exchange of information

Exchange of information on fire safety and fire protection at NPPs were continuing activities. Technical Committee Meetings have been organized regularly every year in order to review draft documents, to obtain advice on the direction and scope of future project activities and to reach international consensus on selected fire safety related topics. In addition, these meetings provided a forum for the exchange of up-to-date information between practitioners on fire safety related issues.

This Symposium is one of the relevant events that facilitate the exchange of technical information. It focuses on effective methods, practices and criteria applied in the fire safety assessment and upgrading of nuclear power plants.

One of the tasks that serve the purpose of information exchange in the area of fire safety assessment is the development of a list of fire safety related technical issues. The utility of a list which represents broader international views and the potential role the IAEA may play in its compilation provide a good motivation to make an effort in this direction. This activity will be discussed in more details in this paper.

2.3. Fire safety reviews and training

Assistance is provided in the systematic assessment of fire safety of nuclear power plants in Member States. The organization of fire safety review missions and training courses on fire safety and fire protection are continuing activities.

Six fire safety review missions were conducted in the period 1993-1997 to evaluate the adequacy of fire safety in selected nuclear power plants. These plants were in different stages of construction and operation and the scope and depth of the evaluation varied. Some practical insights from these fire safety review missions are discussed in the paper entitled "IAEA fire safety review missions to NPPs" (No. 29).

An IAEA document [7] providing guidance for the experts involved in the organization and conduct of fire safety review missions has been completed and approved for publication in IAEA-TEC-DOC Series. A more detailed information on this document is provided in the paper No. 28 presented at this Symposium.

Interregional training courses were typical training activities organized by the IAEA in the area of fire safety. Two interregional training courses on "Fire Protection and Environmental Qualification of Equipment Important to Safety in Nuclear Power Plants" were organized - one in Mumbay (formerly Bombay), India (in 1995), and the other divided between Argonne, USA and Toronto, Canada (in 1996). Several specific training missions or workshops initiated at the request of a Member States have been provided within the framework of Technical Assistance and Co-operation programmes.

3. ONGOING WORK

Current activities in the area of fire safety of nuclear power plants focused on the use of fire related operational experience. There are two IAEA-TEC-DOCs under preparation devoted to this subject: "Use of Operating Experience in Fire Safety Assessment of Nuclear Power Plants" and "Root Cause Analysis of Fire Safety Related Events at Nuclear Power Plants". The main objectives, intended scope and contents of these documents as well as the main insights obtained so far are presented below.

3.1. Use of operational experience in fire safety assessment of nuclear power plants

The objective of this TEC-DOC is to provide good practice information on the collection and use of fire safety related operational experience. The publication is intended to assist plant/utility staff responsible for collecting plant raw data related to fire safety and the analysts involved in fire safety assessment of NPPs.

The material presented in the document may help to make informed decisions related to data reporting systems by the utilities/plants. It is essential that a rational compromise between 'analyst's wishes' and data collector's ability and willingness to provide required input is reached. The document is intended to address the subject taking this aspect into consideration.

The document will describe potential applications of operational experience in the assessment of fire safety for NPPs. Desirable scope of data reporting to satisfy the needs specific to various applications and to various aspects of fire safety assessment will be discussed with special consideration given to the needs of risk based fire safety assessment. The availability and accessibility of information in the existing data reporting practice will be addressed and some practical examples of data reporting systems will be given. The document will also provide some advice on how to use the available nuclear industry data, often imperfect or incomplete, in order to generate acceptable plant specific data.

The preparation of the document is well advanced. However, the existing draft is not complete and has to be extended to cover additional topics mainly related to the application of the existing experience based data. Work on this document is planned to be continued in 1997/98. Some insights from this study regarding the potential applications, related data requirements, current reporting practice and the use of industry data in plant specific analyses are summarized below.

Direct use of operational experience such as root cause analysis or trend analysis is the most straightforward application. These methods are powerful tools capable to identify fire protection deficiencies in safety culture area which could be hardly identified by plant inspection or by quantitative hazard analysis. Direct methods, in particular the root cause analysis, are most effective when conducted at the plant level with strong involvement of plant personnel.

Indirect use of historical plant data is of prime interest to any quantitative fire safety analysis. The main data derived from plant operational experience include fire event frequencies, fire detection and suppression times, probability of fire growth as well as failure rates for human actions and for fire protection equipment.

Use of historical operational data in a qualitative way in order to support a plant specific modelling of physical fire phenomena is another important application. However, there are some practical limitations as the existing fire event reporting systems are not sufficiently detailed to allow for suitable extrapolation to the conditions specific to plant, plant fire compartment and/or the threat being considered in a plant specific analysis.

Direct in-depth analyses such as root cause analysis or trend analysis require information on broad range of events including those that do not have safety significance but are relevant to safety (deviations). The level of detail of information required for these applications (e.g. for identification of root causes) is relatively high. The information scope is usually beyond the scope of a typical fire event reporting system. Information sources may include a large variety of plant operational records (e.g. station log, plant control log, workshop logs, maintenance records), special investigation reports, manufacturing/erection records and interviews with plant personnel involved.

Indirect application of operational experience relay mainly on fire event data. Fire event frequencies, fire detection and suppression times, probabilities of fire growth and failure rates for human actions are estimated using historical fire event information. For reliability rates of fire protection equipment this source of data is not sufficient. Information on the results of surveillance checks, periodic tests and maintenance activities is essential for this application.

The availability and quality of data derived from plant operating experience are recognized unsatisfactory. In some Member States fire safety related events are not reported in systematic and consistent manner. Often the criteria, scope and format of data reporting need improvement. The currently available sources of data have not been properly maintained and are often out of date.

Typical fire event reporting systems have been designed to assist in direct applications of data (root cause analysis, trend analysis, regulatory supervision, etc.). They include mainly safety significant events or events involving significant property loss. Fires of low severity or precursor events are not reported.

In many cases the definition of criteria and format used in data reporting systems are not precise and involve subjective judgment. That negatively affects the reliability of information provided and the understanding of fire event reports and in consequence their usefulness.

Considering the limited availability of statistical plant data on fire events as well as other data for estimating the reliability and performance of fire protection equipment and systems, use of the available worldwide data would be desirable. However, due to the lack of standardized criteria and format, a broader exchange of data at international level and sometimes even at national level is difficult.

The information provided in the existing fire event reporting systems is in many cases unsatisfactory to fulfill the actual needs of fire risk assessment. Information on event time line required for estimating the fire detection and suppression times is limited. In certain cases the performance of the fire detection and fire suppression systems is also difficult to be assessed correctly due to the lack of sufficient information provided for the event. Typically, the scope and level of detail of information provided do not allow for implementing more refined event partitioning techniques to estimate the occurrence frequency of a given fire in a given plant location (e.g. due to the lack of information on the relevant attributes of the related fire compartment).

Enhancement of the current reporting process would be essential aid to future efforts to learn from past fire events. A more precise definition of reporting criteria, scope and format is essential to facilitate exchange of data at national and international level. It is desirable to broaden the scope of data reporting by including events of lower severity and by adding new types of information to be provided for each event. Certain detailed data may be made available in a plant level system and appropriate cross reference provided in the higher level system. For instance, detailed attributes of the related fire compartment (e.g. the presence of ignition sources, fire load and its distribution, detection system and extinguishing equipment) can be treated in this way.

Extrapolating experience data from plants with different level of fire protection defenses is subject to debate and disagreement. One of the important applications in which a broader use of historical data would be desirable is estimating the occurrence frequency of a given fire in a given plant location (fire event partitioning). There is no clear consensus regarding the best approach to fire event partitioning. Fire occurrence models used in the partitioning process are often simplistic and do not take into account all relevant attributes of the fire location. Implementation of more sophisticated approaches is limited by the practical availability of the related plant specific information in the databases.

3.2. Root cause analysis for fire safety related events

The objective of this TEC-DOC is to promulgate the use of the ASSET root cause analysis methodology for application to the analysis of events involving fire. The document is intended to be used by fire safety assessors focussing on the direct use by the plant staff involved in fire protection. Development of this document is well advanced (a first draft has been prepared).

The document presents the ASSET root cause analysis tailored towards its application to the investigation of fires. The methodology is described and illustrated through reference to a fictitious example. The methodology has been applied to three events in which fires were involved. These events are based on real operational experience of three reference plants. The document illustrates both the practical application of the methodology and the nature of the recommendations which arise. Conclusions from these three analyses are presented highlighting the general weaknesses observed.

The three referenced events analyzed had been investigated on the plants without using root cause analysis. The application of the ASSET root cause analysis methodology has extended the insights into the causes of the incidents.

Weaknesses in the field of quality control, surveillance programmes and safety culture were identified in each case and the importance of these weaknesses was shown. In each of the three causes, the event (and all its consequential costs) would have been avoided if only appropriate attention had been paid to these "software issues". It remains only to be said that in general the corrective action in these areas are less costly then the corrective action involving equipment and can usually be put in place in a shorter space of time.

It follows that if the ASSET RCA methodology is used on a routine basis to analyze deviations, including minor deviations, weaknesses in the field of quality control, surveillance and safety culture will be identified in a timely manner and if attended to will significantly reduce the incidence of major events.

3.3. Generic fire related safety issues

One of the tasks carried out at the IAEA in the area of fire risk analysis is compiling a list of technical issues which limit the current capabilities of fire safety analysis. There are currently efforts underway over the world to overcome the existing shortcomings and limitations in fire safety assessment. Co-ordination of activities carried out in this area in different Member States is very desirable and a wider listing of the needs would be a strong step in this direction.

The acceptance by a wide selection of experts will be critical to the credibility of such a list. Therefore, the proposed approach is tailored to solicit a broader opinion.

A provisional list of issues is planned to be prepared at the IAEA. This list will be developed in co-operation with the selected organizations and/or individual experts who have sufficient experience and knowledge in the subject. Information gathered during the implementation of IAEA programme will also be used.

The provisional list when compiled will be sent to recognised organizations/experts to obtain their opinion. In this step the recipients will be asked to provide answers to the following questions:

- 'Is the list complete (if not, what should be added) ?'
- 'What issues are applicable (not applicable) to your organization ?'
- 'What work is underway or planned in this area in your organization ?'
- 'What are the priorities in resolving each of the identified issues ?'
- 'What items are of common interest ?'
- 'What form of collaboration could be proposed in resolving each of the identified issues ?'
- 'Do you know any other organizations or individuals that might be interested to contribute to this list ? If so, please forward the list to them for their response.'

In the second iteration some new items may be added to the list and the updated list would be distributed, basically with the same questions.

The development of a provisional list of technical issues is well advanced. Identified issues are grouped into 5 broader categories:

- Experience based data
- Experimental data
- Modelling of fire behaviour and effects phenomena
- Modelling of fire protection systems
- Other issues related to risk assessment.

The identified issues cover the needs of both deterministic fire hazard analysis and risk based assessment. Each issue is identified by a number, and a short title. A more detailed description is provided to clarify each issue.

Support and valuable assistance in the development of the provisional list of issues and the compilation of a distribution list has been given by Mr. C.B. Ramsey[1] (U.S. DOE), Mr. S.P. Nowlen (Sandia National Laboratories) and Dr. M. Röwekamp (GRS). Sandia Report "Improvement Need Areas for Fire Risk Analysis" [8] has been a very helpful inspiration for the development of this list.

4. FUTURE ACTIVITIES

4.1. Guidelines and good practices

Ongoing work on the preparation of good practice documents related to the use of fire safety related operational experience and to the root cause analysis of fire safety related events (initiated in 1997) will be continued. Preparation of a new document highlighting the lessons learned from fire incidents in NPPs is planned to be started in 1998. This document is intended to compile and summarize the existing studies on the assessment of fire related operational events.

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[1] *Also an initiator of this undertaking*

A second revision of Safety Guide "Fire Protection in Nuclear Power Plants" [9] is planned for the period 1999-2000 as part of the revision of a number of Guides in the NUSS Design Series. The revision is intended to update the contents of the Guide taking into account progress that have been made in the last decade in design and regulatory requirements, in fire protection technology and in related analytical techniques.

The revised Guide is expected to address certain technical issues not covered by the existing Guide such as the recommended extent of fire safety analysis and the role of fire risk assessment in design and licensing as well as its relation to deterministic fire hazard analysis. Another issue of interest is the practical application of the single failure principle to fire protection equipment. Interpretation of this principle in the analysis of fire accident sequences remains controversial and is subject of criticism. The revised Guide is expected to provide a more clear guidance on the application of this principle in the deterministic fire hazard analysis.

4.2. Exchange of information

Technical Committee Meetings are planned to be organized regularly in order to review draft documents and to obtain advice on the direction and scope of future project activities. The next TCM is planned to be held in 1998.

4.3. Fire safety reviews and training

A set of IAEA guidelines and good practice documents covering a broad range of design and operational aspects related to fire safety is nearly complete. Therefore, in the near future the IAEA programme on fire safety will focus on the promotion of practical application of the available guidance. This goal will be achieved by providing to Member States safety advisory services that includes fire safety review missions and training missions.

The fire safety missions will focus on solving plant specific problems. The scope of fire safety missions will be tailored to meet the particular requirements of the plant, taking into account the specific problems, conditions and needs of each NPP. These services will also assist in building up the national self-assessment expertise.

Training courses of regional and interregional type are planned to be organized in the area of fire safety regularly. Typical courses on "Fire Protection and Environmental Qualification of Equipment Important to Safety in NPPs" organized by the IAEA in the past are planned to be modified by separating the subjects of fire safety and the equipment qualification. The next training course devoted to fire safety and fire protection in nuclear power plants is planned to be organized in Moscow in 2000.

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CONCLUDING STATEMENT

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1. INTRODUCTION

The title of the Symposium, "Upgrading The Fire Safety of Operating Nuclear Power Plants" reflects the delegates desire and commitment to continually improve the safety of their Nuclear Power Plants.

Public acceptance of nuclear power is based on the perception that the radiological risks presented by Nuclear Power Plants is small and well managed. However even minor fires with no nuclear safety significance affect this perception and could ultimately threaten the existence of nuclear power programs. Whilst the Symposium has concentrated on the upgrading of Nuclear Power Plants to mitigate the consequences of fires, everyone has recognised the importance of fire prevention.

The papers presented at the Symposium have covered a broad range of topics relating to fire protection and its role in maintaining nuclear safety and have provided a comprehensive view of world-wide developments and practices in analysing fire safety on Nuclear Power Plants, and a simple summary of individual sessions would be insufficient to highlight some of the significant issues which arose.

In concluding the Symposium I have attempted to identify those issues which, from the presentations and discussions, have appeared to be important to the overall achievement of maintaining, and most importantly, improving nuclear safety. Therefore I have concentrated on four topics; Data, Fire PSA, Fire Safety Reviews and the proposed introduction of Performance Based Regulation.

2. DATA

It was recognised that accurate and reliable data provides the basis of both deterministic or probabilistic analyses. A number of data collection initiatives have been described. The motivation behind these initiatives is the need to support the expanding use of Fire PSA and to develop the use of risk-informed, performance based regulation.
Many requirements were discussed, but two have featured prominently, these were the fire initiation frequency and data relating to the reliability and unavailability of both active and passive fire protection systems. The value of data in both areas suffered from the generic problems of inconsistent reporting, incomplete information and the use of different reporting standards.

Several database and data collection initiatives have been highlighted including those from WANO, IPSN, India and the NEIL initiative. Whilst these initiatives had the common goal of identifying and recording fire incidents, it was recognised that each was different, adopting different reporting criteria and recording different types of fire information. This is not a criticism of the databases as each was set up for a specific purpose, but it has made any comparison or unification of the data difficult.

It has been proposed that fire frequency deficiencies can be overcome by expanding the plant data to include generic data from other countries such as the United States, but it is recognised that the origins of the data needed to be understood and compared against the plant specific conditions before its use. However, this approach may not be appropriate for the data relating to the reliability and unavailability of fire protection systems, where differences in manufacturing, testing, inspection and maintenance standards are significant. It was concluded that this problem could be addressed if each Nuclear Power Plant were to compile its own plant specific data. The development of the IAEA's TEC-DOC on the "Use of Operating Experience in Fire Safety Assessment of Nuclear Power Plants" should prove useful in this task.

No matter which source of data is used in analysis, its use is conditional on maintaining the operational practices and standards applicable to the source of data or at least adopting practices that are judged to achieve an equivalent standard. Failure to do this may invalidate its use.

There would appear to be a need for international collaboration to improve the data available for fire hazard analysis and this was reflected in delegates request for an internationally accepted fire classification, incorporated into either the IAEA's INES or IRS.

3. FIRE PSA

The considerable interest in Fire PSA is reflected in the number of papers presented. Significant advances have been made over the years in the application of probabilistic assessment to fires, and many of these advances have recently been recorded in the IAEA's Safety Report on Fire PSA.

It was reported that Fire PSA has been used to supplement the deterministic fire hazards analysis and that it is recognised as a tool that has the potential to provide valuable insights with respect to weaknesses in the plant design and operation. It allows the identification of dominant risk contributors, the comparison of options for risk reduction and provides a basis for cost benefit analysis.

Ideally the results of the Fire PSA should enable the fire safety engineer to focus on upgrading those aspects of fire safety which contribute to the greatest risk reduction. The results of the Fire PSA may also indicate where it is not reasonably practicable to improve safety further.

However, some delegates drew attention to current limitations on the use of Fire PSA. These included, incompleteness of data, inadequately conceived modelling or mistakes in the screening out of low frequency events. These could produce fictitious results which may obscure the true fire risks, and create a false sense of security. Therefore the application of the method needed care and detailed knowledge of the fire phenomena and its potential impact on nuclear safety systems. Moreover the scarcity of appropriate data introduces further uncertainty into the results. A number of initiatives are now taking place throughout the world to address these limitations and uncertainties and it is expected that these will feature prominently in the IAEA list of technical issues.

The message appeared to be that the Fire PSA methodology should provide valuable insights into risk contributors but should be used with caution. Because the results of Fire PSA are not testable and are dependent on the analysts knowledge, they should be regarded as input to a decision making process, and judgements on the adequacy of safety levels should also be dependent on compliance with engineering and deterministic safety standards.

In the future, as the utopia of a Fire PSA with no uncertainties is approached, it was anticipated that the technique could be used to refine the deterministic rules to reduce unnecessary levels of conservatism. In the meantime, the use of Fire PSA should not undermine the defence-in-depth strategy or the deterministic engineering approaches that are presently the foundation of regulatory decisions. It should be used in combination with the deterministic methods in a constructive way to identify further cost effective improvements.

4. FIRE SAFETY REVIEWS

The radiological effects of major accidents at nuclear power plants do not respect international borders, and operators of Nuclear Power Plant wherever they are in the world recognise that the risks associated with their Nuclear Power Plants must be adequately managed. It was recognised that it is important to ensure that the management systems and hardware systems, put in place to achieve an adequate level of nuclear safety, are themselves adequately maintained throughout the plants operation.

One of the most effective means of achieving this objective is through periodic fire safety reviews. Some licensing regimes, recognising its importance, require periodic safety reviews and plant and equipment upgrades as technology and knowledge advances and so continually drive risks down. Several examples of the advantages and benefits of independent peer reviews have been presented. Independent peer reviews have been carried out by organisations such as WANO and the IAEA. The Hartlepool Nuclear Power Plant in the UK is currently being peered reviewed by WANO.

A major benefit of such reviews was the bringing together of experts to share knowledge and experience of best international practice with the operating plant personnel and with each other. It was agreed that the scope of the review should be broadened to include an assessment of the management and organisational structure and should not simply concentrate on fire hardware such as fire detection and suppression, as managerial deficiencies have been identified from previous reviews as a significant contributor to the degradation of fire safety.

Self assessment is also possible, and the three IAEA Safety Practices covering different types of fire review should prove useful in this task.

5. PERFORMANCE BASED REGULATION

There was a lengthy debate on the recent USNRC proposal to introduce the principles of risk-informed, performance based regulation of fire protection. The objective of the initiative is to reduce the regulatory burden on licensees by introducing flexibility in fire safety provision in areas where compliance with the prescriptive rules appear to have a marginal affect on nuclear safety. There was general agreement that this approach could be useful.

It was noted that the use of a performance based methodology may allow more cost effective solutions to be developed to the management of fire risk. Consequently, it was agreed that this approach would be more flexible, would allow a quicker response and allow advances in knowledge and technology to be readily adopted. However, it would place a greater reliance on the use of judgement and discretion in the identification of fire safety goals and the development of performance criteria for hardware. A performance based approach also requires validated analytical tools for studies of fire and smoke development and spread.

This approach was not new to many of the delegates as a performance based concept has been applied in considering exceptions to current prescriptive rules.

Most countries, particularly those in the European Community are now adopting a watching brief on the US initiative and are not currently considering any amendments to their current licensing regulations.

6. CONCLUSION

The papers presented at the Symposium covered many and varied topics. They have provided a valuable source of information on safety issues relating to the upgrading of fire safety on operating Nuclear Power Plants, and have described the many upgrading initiatives taking place throughout the world. While this concluding statement has aimed to highlight some of the more significant safety issues, the substance and value of the Symposium is recorded in the detailed papers and in the records of the discussion panels, and these should prove useful to the IAEA in developing it's future programme.

There was obviously a need to address many topics in the advancement of fire safety of Nuclear Power Plants, and many of these issues were identified during the Symposium. Some of them may only be resolved through international collaborations, and the delegates looked to the IAEA to help co-ordinate these.

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Session 3	M. KAERCHER	France
Session 4	M. RÖWEKAMP	Germany
Session 5	H.P. BERG	Germany
Session 6	V.I. POGORELOV	Russian Federation
Session 7	R. WITTMANN	Germany
Closing Session	M.J. KULIG	IAEA

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