FIRE RISK ANALYSIS FOR NUCLEAR POWER PLANTS

Nathan O. Siu and Nicholas B. Melly
U.S. Nuclear Regulatory Commission
Washington DC, 20555
Nathan.Siu@nrc.gov; Nicholas.Melly@nrc.gov

Steven P. Nowlen
Sandia National Laboratory
Albuquerque, NM 87185
spnowle@sandia.gov

Mardy Kazarians
Kazarians & Associates, Inc.
Glendale, California 91207
mkazarians@kazarians.com

Table of Contents

Figures

1. Introduction

Fire risk analysis for nuclear power plants, as currently performed in the U.S. and abroad, is focused on assessing the likelihood of a particular industrial accident: the loss of cooling to the reactor core and subsequent core damage.* This is one main area nuclear power plant fire protection differs from many industrial industries. The main focus on nuclear fire protection is to mitigate reactor core damage events which can be caused by fires within the plant. Nuclear power plants have numerous safety systems intended to prevent or mitigate such accidents, thus the fire risk analysis concentrates on assessing the potential effects of fires on these systems' equipment (components and cables). The effects of fire on building occupants (the plant operators in this case) or the building itself, are factors in the analysis, but are addressed only to the extent that these effects can contribute to the accident.

The nuclear power industry uses a general risk analysis method, known as Probabilistic Risk Assessment (PRA), to estimate the risk associated with all recognized threats to the plant. These threats include equipment failures, human errors, extreme weather conditions, seismic events, and fires within the plant. This chapter addresses fire PRA, i.e., PRA conducted specifically for internal fires.

As discussed by Kaplan and Garrick[1] and in this handbook's Chapter "Introduction to Fire Risk Analysis," risk analysis is the process of:

(1) Identifying potentially important accident scenarios ("what can go wrong"),
(2) Determining their consequences ("what can happen when something goes wrong"), and
(3) Assessing their likelihood ("how likely is it that something will go wrong")

PRA is a form of risk analysis in which both the consequences and likelihood are expressed in quantitative terms. In particular, the likelihood is expressed in terms of mathematically defined probabilities.

PRA has been used in all U.S. plants and most international plants to evaluate the risk associated with electric power generation from nuclear power plants since the landmark "Reactor Safety Study" (*WASH-1400*) published in 1975[2]. PRA, sometimes called "probabilistic safety assessment" (PSA) in international reports, provides a systematic, multidisciplinary approach for using a wide range of information sources including model predictions, experimental results, and plant operational experience, including reported event data, to assess plant behavior under a variety of conditions, sensitivities, and areas of uncertainty and importance.

---

* From a public health and risk standpoint, the integrity of the nuclear fuel is the main safety concern at nuclear power plants. Absent sufficient cooling, the heat generated from radionuclide decay can lead to fuel melting and the potential release of radioactivity into the environment. Past studies have shown that accidents involving the loss of cooling to the reactor core (see Figure 1) are the dominant contributors to risk, and these accidents continue to be the focus of current risk studies. Recent analytical studies, as well as the March 11, 2011 accident at the Fukushima nuclear power plant in Japan, have indicated that accidents involving used ("spent") nuclear fuel outside of the core may be more risk significant than previously thought. Studies to re-assess the potential importance of these accidents are ongoing.

Numerous fire PRA studies have shown that fire can be a significant or even dominant contributor to the overall risk for a given nuclear power plant. Fire-induced Core Damage Frequency (CDF) range from 4E-8 to 2E-4 per reactor year, the majority lies between 1E-6 and 1E-4. Fire events are not a design basis accidents considered by the nuclear industry; yet the risk they pose to core damage frequencies has been reported to exceed 10% of internal events CDF for most Nuclear Power Plants (NPP). In some cases, 25% of operating reactors, report that fire-induced CDF even exceeds corresponding internal events CDF. Lessons learned from a number of serious events, including Browns Ferry (United States, 1975),[3] Armenia (Armenia, 1982),[4] Vandellos (Spain, 1989),[5] and Narora (India, 1993),[6] further emphasize fire's potential importance. NUREG/CR-6738, "Risk Methods Insights Gained from Fire Incidents," provides a useful, PRA-oriented review of these and other notable nuclear power plant fires.[7***]

Whether fire is an important risk contributor at a particular plant is determined by, not only the reactor design, but also such plant-specific differences as locations for redundant, diverse safety equipment, routing of key electrical cables (e.g., the separation and orientation of the respective cable trays and conduit), fire protection schemes for particular rooms, and the procedures employed by plant operators in response to a fire. Fire PRA evaluates these details and shows how they relate to risk. It provides a systematic framework for examining the complex phenomenology underlying a fire using a wide variety of information sources (e.g., experimental results, model predictions and reported event data), and furnishes a useful context for discussions of areas with significant controversies and uncertainties.

Fire PRA (and PRA in general) is performed to support decision making. The decision problems can be faced by the plant owner (e.g., how to rationally allocate safety resources) or by the regulator (e.g., whether to accept a proposed plant change). The increasing, more direct use of fire PRA in regulatory applications, which is consistent with the NRC's PRA Policy Statement,[8†] implies a need for a high level of fidelity. It is important that the fire PRA be sufficiently realistic to appropriately address the decision problem at hand. Both excessive optimism and excessive conservatism need to be avoided, as biased analyses could lead to suboptimal or even inappropriate decisions. The problems associated with an overly optimistic analysis are clear. On the other hand, conservatism has traditionally been viewed as, in effect, the price paid for imperfect knowledge. Common engineering practice is to apply realistic assumptions wherever possible, but when in doubt, conservative assumptions generally prevail. However, excessive or uneven levels of conservatism might lead to inappropriate conclusions. For example, an excess of conservatism in the analysis of one particular set of fire sources may cause the fire risk for those sources to be sharply over-estimated. This might, in turn, mask the importance of another risk contributor where more realism and less conservatism have been applied.

Numerous papers and reports have been written on fire PRA. The objective of this chapter is

---

[**] The U.S. Nuclear Regulatory Commission (NRC) designates its staff-prepared reports using the nomenclature "NUREG." NUREG/CR contractor reports are reports prepared by NRC contractors.

[†] Among other things, this statement indicates that the NRC intends to increase its use of PRA technology "in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data…".

to provide a general review of the subject as it applies to NPP. The chapter presents some key characteristics of fire PRA, discusses the fire PRA methodology employed by most current domestic studies, summarizes results of a number of analyses, and then briefly outlines current activities and anticipated future developments. Detailed guidance needed to perform a fire PRA can be found in cited references, notably NUREG/CR-6850/EPRI TR-1011989,[9‡] a fire PRA methodology report jointly developed by the Electric Power Research Institute (EPRI) and the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES). Recognizing that fire PRA is a multidisciplinary enterprise, this chapter touches on all elements of the fire PRA but the emphasis is placed on topics of direct interest to fire protection engineers. Details on those aspects of fire PRA requiring input from other disciplines, including electrical engineering, human factors, and nuclear power plant systems analysis can be found in NUREG/CR-6850/EPRI TR-1011989 and other guidance documents cited in this chapter.

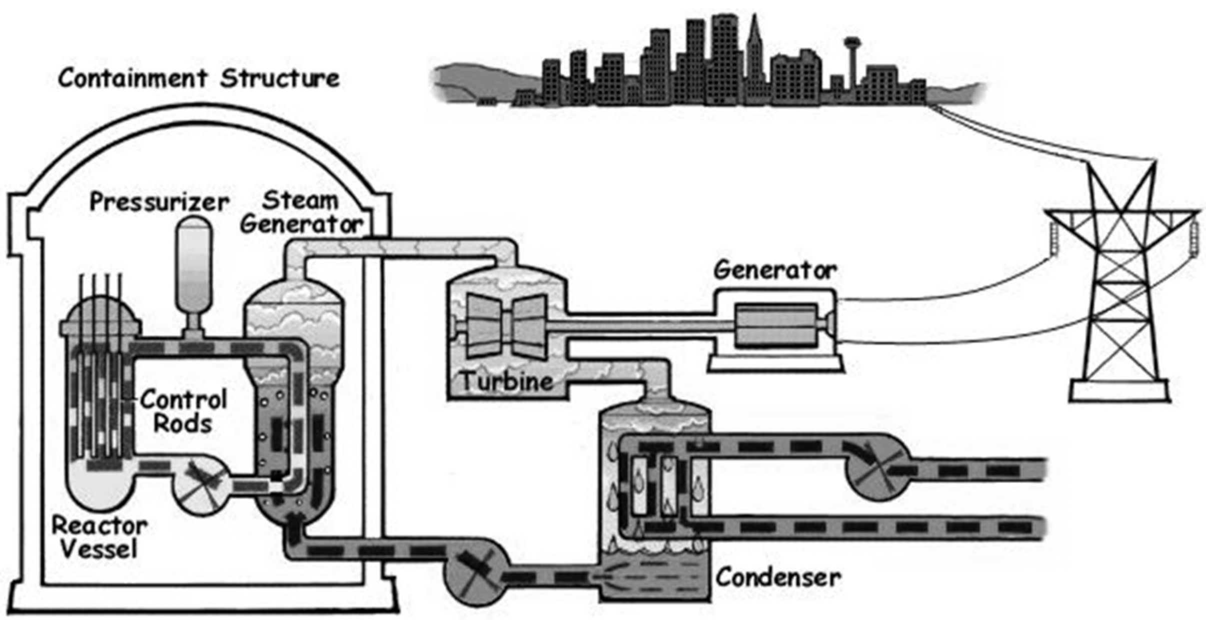

Figure 1- Pressurized Water Reactor (PWR) Nuclear power plant schematic overview

‡ NUREG/CR-6850/EPRI TR-1011989 will be referred to as NUREG-6850 for simplicity

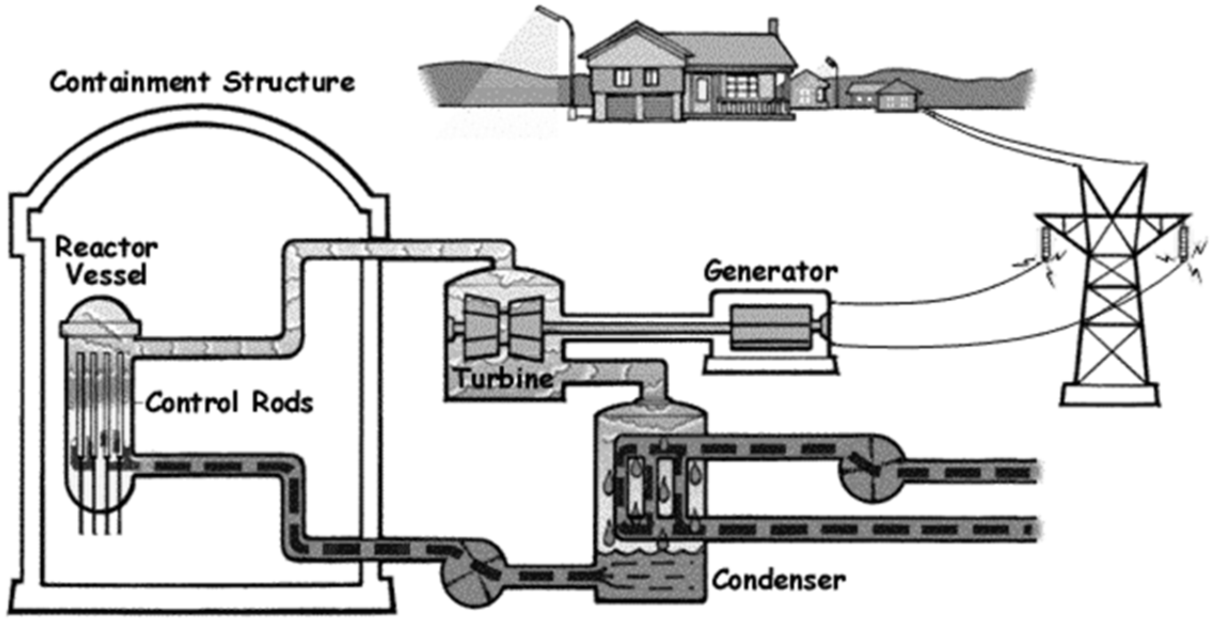

Figure 2- Boiling Water Reactor (BWR) Nuclear power plant schematic overview

2. Nuclear Power Plant PRA and Fire PRA

Before discussing some of the details of fire PRA, it is useful to note some of its key characteristics and history.

First, in most applications of interest, the risks of eventual concern involve the health of the general public. Thus, the focus of the analysis is on accidents that can have significant health effects offsite. Fire PRA methods can be applied to plant worker safety and economic issues, but have not been to date.

Second, rather than directly estimating the likelihood of offsite health effects, fire PRA studies typically are aimed at assessing a surrogate measure of risk, the plant "core damage frequency" (CDF). This measure characterizes the likelihood that the plant will suffer an accident that damages nuclear fuel contained in the reactor core.[§] As discussed previously, the attention on core damage is due to the results of past studies (e.g., WASH-1400) that have shown that core damage accidents are the dominant contributors to overall plant risk. It is important to recognize that nuclear power plant fires, by themselves, will not lead to core damage; that is, plant fires do not directly threaten core integrity. In order to cause core damage, a fire-initiated scenario must involve a plant upset condition (i.e., an accident scenario "initiating event") and the compromise of plant safety systems. Damage to plant systems may occur either directly (e.g., fire-induced damage to a pump) or indirectly (e.g., fire-induced damage to supporting equipment such as electrical cables or power distribution busses).

Third, fire PRAs usually focus on accident scenarios involving fire-induced failure of critical electrical cables and cabinets. The notion of "critical" arises from the degree of redundancy and defense-in-depth built into nuclear power plant designs. In practice, a risk significant fire must actually damage most or all of the safety systems provided to handle the initiating event. Multiple, independent safety system failures due to random causes (i.e., non-fire causes), while possible, tend to be far less likely and therefore less important to overall risk. Risk significant fires typically involve multiple system failures caused by fire-induced damage to collocated electrical cables (including instrument & control cables) that support these systems. In fact, one of the most predominant fire safety strategies in nuclear power plant applications is ensuring physical separation (or passive fire protection) for important electrical cables.

Fourth, fire PRA is normally performed following, or in concert with, the performance of a PRA addressing accident scenarios initiated by hardware failures or operator errors. Such a PRA, which is called an "internal events" PRA for historical reasons, includes detailed event tree and fault tree models that identify potential responses of the nuclear power plant and operators to postulated initiating events. (Figures 2 and 3 provide abbreviated examples.) In the fire PRA, these internal event models are modified to address those events and conditions introduced by potential fires. It is important to recognize that the set of modified models, called the "plant response model" later in this chapter, needs to include the effects of both fire- and non-fire caused equipment failures and human errors.

[§] Technically, the CDF is the expected (in a statistical sense) number of core damage events per unit time.

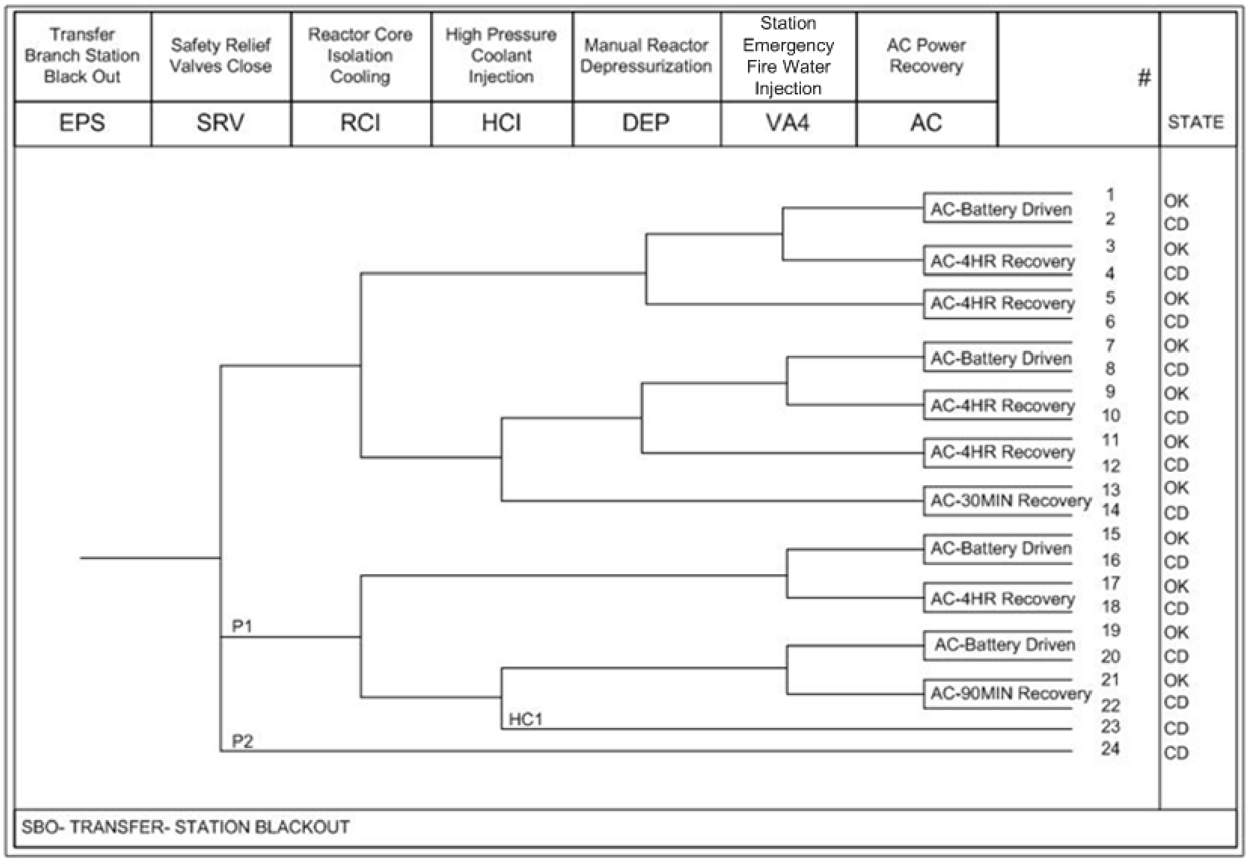

Figure 3- Example event tree for internal events analysis

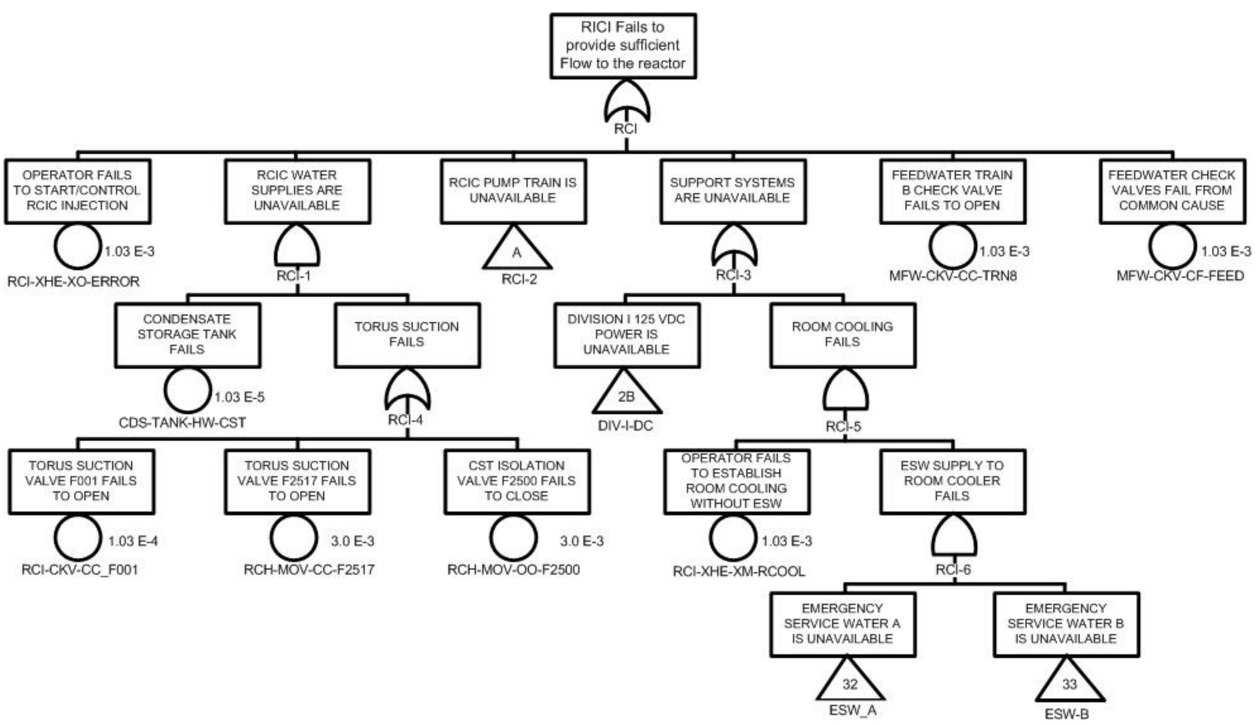

Figure 4- : Example portion of a fault tree for internal events analysis (model used depicts a Boiling Water Reactor BWR)

Fifth, those fire PRA studies that quantitatively treat uncertainties do so within the methodological framework employed by the overall plant PRAs. Consistent with this handbook's chapter "Uncertainty and Safety Factors," and as discussed later in this chapter, this framework distinguishes between aleatory (also called random or stochastic) uncertainties and epistemic (also called state of knowledge) uncertainties.[10] The distinction between aleatory and epistemic uncertainties is useful for decision making because it helps identify appropriate risk management options. For example, aleatory uncertainty in the occurrence of core damage events can be reduced by making design changes to reduce CDF, whereas epistemic uncertainty can be reduced through research on key phenomenology (e.g., electrical circuit behavior when cable bundles are exposed to fires). NUREG-1855[11] describes the treatment of uncertainty in a decision support context.

Sixth, because fire PRAs often need to consider the risk contributions of an enormous number of scenarios, these analyses are almost always performed in an iterative fashion. Starting with a fairly coarse analysis using conservative modeling assumptions, the analysis team develops an understanding as to what scenarios are likely to be important contributors to overall risk. Focusing attention on important scenarios, the team progressively adds detail and more realistic modeling approaches until the analysis is sufficiently accurate to support the decision problem being addressed by the fire PRA. This approach does not guarantee an accurate assessment of absolute CDF (or risk). A conservative assessment that is adequate for one decision problem (e.g., whether additional regulatory action is needed to address fire vulnerabilities in the nuclear fleet) may be overly conservative with respect to a different decision problem (e.g., selecting between different plant-specific risk management solutions).

We note that certain situations that have received little or no coverage from general PRA studies also receive little or no coverage in the fire PRA. These situations include the scenarios associated with acts of sabotage and scenarios when the plant is not operating at steady-state, full power conditions. Sabotage is usually excluded from the scope of general PRA studies. Regarding scenarios during low power or plant shutdown operating conditions, fire PRAs have been performed but[12, 13] such assessments are not yet routine. Draft NUREG/CR-7114[14] provides a framework for performing fire PRA for these conditions.

3. History of Fire PRA

The earliest fire risk assessment for a nuclear power plant was performed in 1975 as a supplement to WASH-1400 (the Reactor Safety Study).[2] The assessment was aimed at providing a quick estimate of the risk implications of the Browns Ferry cable fire in 1975. The analysis indicated that the CDF associated with that fire was around 10^{-5} per year, or about 20 percent of the total plant CDF associated with the accidents addressed in the main body of the study. It also noted the usefulness of developing a more detailed fire PRA methodology (including improved models and data). Another early fire PRA was performed in 1979 as part of a PRA for a proposed high-temperature gas-cooled reactor design. The analysis focused on the risk contribution of cable spreading room fires, and it concluded that the core heat-up frequency due to such fires was also around 10^{-5} per year, or about 25 percent of the total core heat-up frequency due to all causes.

The first comprehensive, detailed fire PRAs for commercial nuclear power plants were performed in 1981 and 1982 as part of the commercially-sponsored Zion[15] and Indian Point[16] PRA studies, respectively. A key question addressed by both PRA efforts was if additional accident mitigation systems (e.g., filtered, vented containments) were needed for the two plants. The study results indicated that the fire risk for Zion Units 1 and 2 was relatively small (the mean CDF was about 5 x 10^{-6} per year for each unit, about 10 percent of the total mean CDF) and that the fire risk at the Indian Point plants was relatively large (the mean fire-induced CDF for Unit 2 was about 2 x 10^{-4} per year, about 40 percent of the total mean CDF). Because the Zion and Indian Point fire PRA studies were performed by the same analysis team using the same analysis methodology and tools, these studies demonstrated how plant-specific features could greatly affect fire risk. More important, the studies also identified plant design changes for reducing risk (e.g., fire barriers, a self-contained charging pump, provisions for an alternate power source in the event of damaging fires) that were assessed to be more cost-effective than the proposed accident mitigation systems prompting the studies.

In the years following the Zion and Indian Point studies, a number of additional fire PRA studies were performed. The results of these analyses confirmed that fire could be a significant and even dominant contributor to the overall risk for a given plant. Many of these studies resulted in estimates of mean fire-induced CDFs of 10^{-4} per year or greater, predicted contributions to total CDF of 20 percent or greater, or both. In the late 1980's through very early 1990's, the NRC sponsored a Risk Methods Integration and Evaluation Program (RMIEP)[17] at the LaSalle Unit 2 NPP and NUREG-1150[18,19] PRA studies, each of which included assessments of fire risk. The RMIEP study involved an analysis of the LaSalle nuclear power plant and the goal of the study was to extend and demonstrate state of the art analysis methods for PRA in general. The NUREG-1150 studies focused on the application of existing methods, rather than the development of new methods, to a set of five reactors representing a range of U.S. designs. From the standpoint of fire PRA, both studies used similar frameworks and (for the most part) methods. Insights from both of these programs paralleled other PRA studies of the era and again found that fires were potentially significant contributors to overall plant risk. Both studies also confirmed that plant-specific details can substantially impact both the magnitude and source of fire risk.

In 1991, recognizing the value of systematic assessments of fire (and other so-called "external events"), the NRC requested that licensees perform Individual Plant Examination of External Events (IPEEE) studies for their plants.[20]The primary goal of the fire risk portion of the IPEEE program was for plant licensees to identify plant-specific vulnerabilities to fire-induced severe accidents that could be fixed with low-cost improvements. Four supporting objectives with respect to external events were for licensees to (1) develop an appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur under full-power conditions, (3) gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) reduce, if necessary, the overall likelihood of core damage and radioactive material releases by appropriately modifying hardware and procedures to prevent or mitigate severe accidents. Guidance on the performance of an IPEEE analysis was provided by Chen et al.[21]

The results of the NRC's review of the IPEEE submittals are presented in NUREG-1742.[22] The review showed that fire was a significant contributor to overall risk at a number of plants. In addition, it also showed that over half of the IPEEE submittals identified cost-effective improvements. Finally, as discussed earlier in the section on fire modeling, the review process also identified a number of technical issues associated with then-current guidance.

In the early 2000s, as the IPEEE review process was coming to a close, EPRI and the NRC's Office of Nuclear Regulatory Research initiated a joint effort to update available fire PRA methods and guidance. This effort was intended to consolidate lessons learned from the IPEEEs and from ongoing fire PRA research and development activities, and was aimed at supporting anticipated regulatory applications of fire PRA (discussed in the following section). This joint effort resulted in the guidance document NUREG/CR-6850, cited extensively in this chapter.

Currently, a number of U.S. plants are using NUREG/CR-6850 and more recent guidance[23] to update their fire PRAs. These efforts have resulted in a number of lessons learned, including the following.

- As in previous fire PRAs, the fire risk profile is often dominated by the contribution from a relatively small number (on the order of ten) scenarios. These scenarios often involve situations identified in past studies (e.g., electrical cabinets, control rooms, locations with large concentrations of cables) with major consideration given to targets vulnerable to fire-induced spurious operations. These targets can have a pronounced impact on the fire scenarios selected.
- Overall, current estimates of fire CDF (with mean values on the order of 10^{-5} to 10^{-4} per year) are consistent with the results of past studies. Note that some of these estimates reflect the assumption that planned modifications are in place and operational.[24]
- NUREG/CR-6850 provides guidance on performing screening-level (intentionally conservative) analysis as well as guidance for detailed (realistic) analysis. Fire PRAs that do not employ the latter guidance can produce unrealistically conservative fire CDF estimates.
- Although the fire CDF tends to be dominated by the contribution from a few scenarios, the total fire CDF is also a function of the contributions from other scenarios. A realistic fire PRA can require the quantitative analysis of large number (on the order of a few thousand) scenarios. Such a study requires considerable resources to trace cables as well as perform analyses.[24]
- Fire PRA is a useful tool for identifying plant modifications that will improve plant safety with respect to non-fire as well as fire-initiated accidents.

The recent updating efforts have also identified areas where additional improvements are needed to increase the realism of results and to reduce the effort needed to perform a realistic fire PRA. These areas are the subject of current fire PRA research and development activities. The results of these activities are aimed at supporting the continued use of fire PRA in ensuring and, as needed, improving the fire safety of U.S. nuclear power plants.

Although this chapter is centered on U.S. plants and practices, we note that fire PRA is also widespread internationally. In particular, the Organization for Economic Cooperation and

Development's (OECD) Nuclear Energy Agency (NEA) has published the results of a member country survey on their PRA activities. This report and its more recent update indicate that fire PRAs have been performed for a large number of nuclear power plants abroad.[13,25] The OECD/NEA has also published a state-of-the-art report[26] and supports a number of activities aimed facilitating the exchange of fire-PRA relevant information among member countries. Guidance on the performance of fire PRA is also available from the International Atomic Energy Agency.[26],[27] In general, current international guidance and practices are generally consistent with those in the U.S., and the results of international studies are also qualitatively consistent regarding the relative importance of fire (as compared with other hazards) and the plant areas that are important contributors to risk.

4. Fire PRA Guidance and Standards

Fire protection for the U.S. commercial nuclear power industry is governed by several NRC-issued documents. The primary regulations are published in Title 10, Section 50.48 of the Code of Federal Regulations (10CFR50.48).[28] These primary regulations are supported by a number of subsidiary regulatory guidance documents. Of particular interest to fire PRA are Regulatory Guides (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"[29] and 1.200, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities."[30] These documents, respectively, provide one acceptable method to perform risk-informed, performance-based fire protection programs; and on quality expectations for the use of general PRA (including fire PRA) in regulatory decision making. Also of interest are RGs 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,"[31] and 1.189, "Fire Protection for Nuclear Power Plants."[32] The former provides a framework for the application of PRA information in the regulatory context, and the latter addresses specific fire protection issues. Links to current fire protection regulation, guidance, and other regulatory documents can be found on the NRC's fire protection website[**].

The current guidance for performing a fire PRA is provided in NUREG/CR-6850 and a series of companion documents covering various methods refinements, clarifications, and expansions developed since that publication was released[23].[††] NUREG/CR-6850 is built on lessons learned from the performance and review of past fire PRAs, including the NRC sponsored NUREG-1150 studies[18] and the industry methods and corresponding studies performed as part of the NRC's Individual Plant Examinations of External Events (IPEEE) program.[21,22,33,34] The document also reflects the results of research and development activities (e.g., regarding the treatment of fire-induced failure of cables and circuits) that led to improvements in the treatment of specific fire PRA issues, as well as lessons learned from the application of a draft form of the document in a number of field tests. Similarly, fire PRA guidance developed after NUREG/CR-6850 have

[**] http://www.nrc.gov/reactors/operating/ops-experience/fire-protection.html

[††] Guidance documents on a number of specific fire PRA issues can be found on the NRC public web site. Agency documents are maintained on its Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html#web-based-adams. A number of fire PRA methodology enhancements and clarifications have been developed as a part of industry and NRC efforts to implement risk-informed performance-based fire protection based on the NFPA-805 standard. These can be found using the search term "NFPA 805 FAQ".

benefitted from industry applications of that report. Fire PRA will continue to evolve, and it is often through applications that refinement needs are identified. Research is also ongoing. As research results are developed, these are factored into the fire PRA methods and guidance as appropriate. Methods development and related research activities are discussed later in this chapter.

A joint American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) consensus standard providing requirements for nuclear power plant PRAs quality and scope, including fire PRAs, is also available.[35] This standard delineates the requirements (i.e., the "whats") for a quality fire PRA, but does not prescribe particular methods (i.e., the "hows") for achieving these requirements. The discussion of methods is left to guidance documents, such as those discussed above. A second national consensus standard, National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," provides requirements for developing and maintaining a risk-informed, performance-based fire protection program.36 The 2001 edition of NFPA 805 has been endorsed by the NRC (10 CFR 50,48(c)) as an optional alternative to the deterministic regulatory requirements in place since 1980.

As noted above, PRAs have historically focused on at-power plant conditions (i.e., with the plant up and running at full power). This is reflected in the cited methodology documents and implementation standards. For example, NFPA 805 focuses on the use of fire PRA as a tool supporting fire protection decision making under at-power plant conditions. This standard does require consideration of low-power and shutdown (LPSD) risk, but current decision making relies mainly on qualitative, defense-in-depth methods. Defense in depth is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures. It is envisioned that LPSD applications may evolve to be more quantitative. Standard approaches for performing LPSD PRAs are in the developmental stage. NUREG/CR-7114[14] presents a PRA method framework for quantitatively analyzing fire risk in commercial nuclear power plants during LPSD conditions.

Figure 4 illustrates the relationship between various classes of documents and specific documents that govern the application of risk information to nuclear power plant applications. The top tier represents the actual regulations that govern plant operations as published in the U.S. Code of Federal Regulations (CFR). For example, NRC's endorsement of the NFPA 805 standard is found in 10CFR50.48(c). The second tier represents various regulatory guides (RG) which are documents prepared by the NRC staff that generally describe one acceptable method (non-exclusive) for meeting the regulatory requirements. (Four relevant RGs were cited above.) Also represented in the fourth tier is a Nuclear Energy Institute (NEI) guidance document (NEI 07 12[37]) which provides specific guidance on implementing the ASME/ANS standard. Finally, the fifth tier represents specific methodology guidance documents that express how to perform various aspects of a risk study and includes NUREG/CR-6850 which is heavily cited in this chapter. Tiers 3-5 are all associated with PRA quality expectations: that is; these tiers together define quality expectations for analyses that are to be used in the regulatory decision making

process.

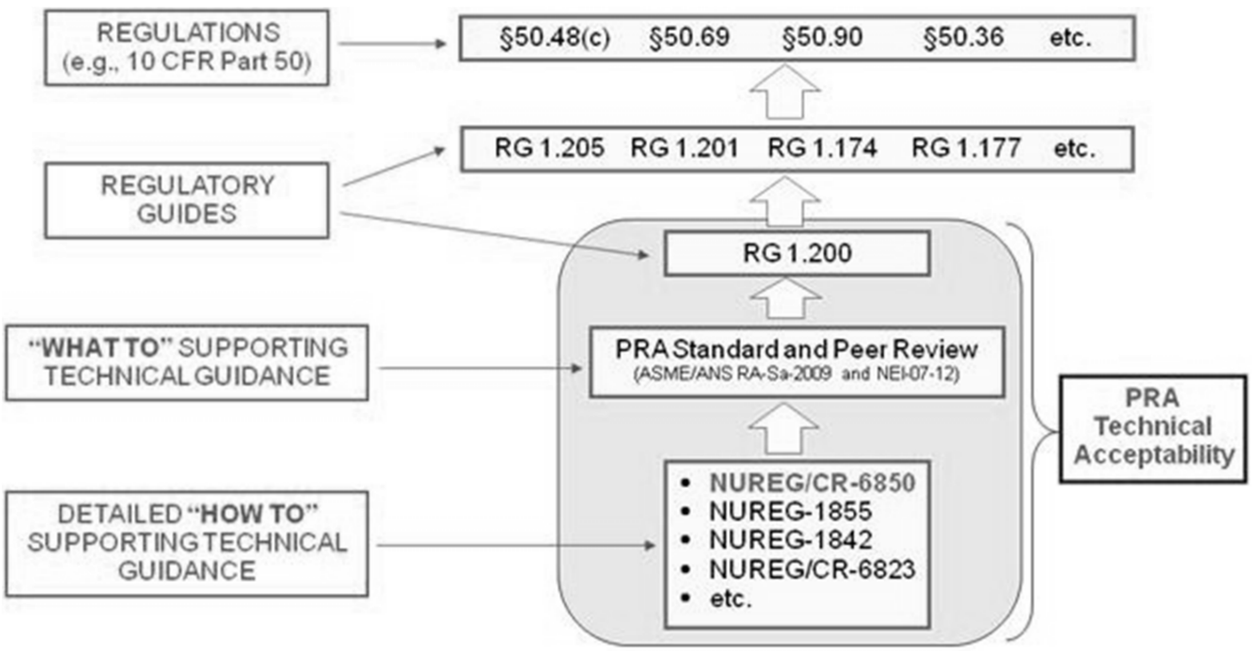

Figure 5- Document hierarchy associated with nuclear power plant fire risk regulatory applications

5. Fire PRA Overview

The fundamental structure of current fire PRAs is little changed from that used in the Zion[38] and Indian Point[39] PRA studies of the early 1980s and described by Apostolakis et al.,[40,41] a structure that is also called out in the 1983 PRA Procedures Guide.[33] In this structure, the CDF arising due to fire-initiated accidents is the sum of the CDF contributions from individual fire-initiated scenarios contributing to this total CDF. The CDF contribution due to a single fire scenario (where, in this discussion, a fire scenario is defined by the location and burning characteristics of the initiating fire), in turn, can be divided into three principal components:

1. Frequency of the fire scenario
2. Conditional probability of fire-induced damage to critical equipment given the fire
3. Conditional probability of core damage given the specified equipment damage

Mathematically,

$$CDF_i = \lambda_i \times p_{ed,j|i} \times p_{CD,k|i,j} \tag{1}$$

Where:

λ_i = Frequency of fire scenario *i*

$p_{ed,j|i}$ = Conditional probability of damage to critical equipment set ("target set")‡‡ *j* given the occurrence of fire scenario *i*

$p_{CD,k|i,j}$ = Conditional probability of core damage due to plant response scenario *k* given fire scenario i *and* damage to target set *j*

The first term accounts for the likelihood that a fire of specified characteristics will occur in a given location within the plant. The second term addresses the likelihood that the fire will not be suppressed prior to component damage (this term may include a measure for the severity of the fire). Analysis of this term requires treatment of the issues of fire growth, detection, suppression, and component damageability. The third term, often referred to as the Conditional Probability of Core Damage (CCDP), addresses the ability of the plant to achieve safe shutdown given the loss of equipment damaged by the fire. Analysis of this term also requires treatment of the unavailability or random failure of equipment unaffected by the fire and the treatment of potential errors by plant operators.

The total fire CDF is obtained by summing Equation 1 over all possible fire scenarios (index i), target sets (index j), and plant responses (index k):

$$CDF = \sum_{i} \lambda_i \left[\sum_{j} p_{ed,j|i} \left(\sum_{k} p_{CD,k|i,j} \right) \right] \tag{2}$$

The three-term breakdown used in these equations provides a useful discussion framework because it aligns with the objectives of fire protection defense-in-depth for nuclear power plants, as discussed in RG 1.189. Three objectives of defense in depth are: first to prevent fires from starting; if that fails, then to detect rapidly, control, and extinguish those fires that do occur; and if that fails to provide protection for structures, systems, and components important to safety so that a fire not promptly extinguished by fire suppression activities will not prevent safe shutdown of the plant. Further, the breakdown aligns with the modeling approaches used in different parts of a fire PRA. In particular, the fire frequencies are generally estimated using simple statistical models for fire occurrences, the likelihood of fire damage is estimated using combinations of deterministic and probabilistic models for the physical processes involved, and the likelihood of core damage is estimated using PRA event tree and fault tree models (modified appropriately to address the needs of fire PRA). The current approach for addressing each of these technical elements is more fully discussed later in this chapter.

‡‡ "Targets" in this context are PRA components and cables near the fire location and "target sets" are simply collections of individual targets that may be threatened by a given fire.

6. Fire PRA Process

Figure 5 provides a schematic of the overall process for performing a fire PRA. This figure is based on the technical elements set forth in Section 4 of the ASME/ANS PRA Standard, which closely parallels the process defined in NUREG/CR-6850.[§§]

As discussed in greater detail in the remainder of this section, the fire PRA process identifies and characterizes, for each plant area, potentially important fire scenarios, installed fire protection systems and features, physical and geometric characteristics required to support fire modeling, and potentially important equipment, components and cables (i.e., nuclear-safety relevant items potentially susceptible to fire damage). The Plant Familiarization elements shown in Figure 5 establish the boundaries of the analysis and define the key characteristics of the plant in both physical and systems modeling contexts. Qualitative screening is then done to identify locations and scenarios that need not be quantitatively analyzed because they are not expected to contribute to the fire CDF. This first screening step typically eliminates plant areas that meet the following criteria: (1) the area houses no potentially important equipment, components, or cables and (2) a fire in the area would not be able to initiate a plant upset condition (i.e., an accident scenario "initiating event").

Following qualitative screening, the fire PRA continues with Detailed Fire Analysis and Plant Response Analysis elements. These elements involve a series of progressively more detailed quantitative assessments for unscreened plant areas. Conservative quantitative analyses are usually performed in initial evaluations of CDF to identify additional plant areas and scenarios that can be screened from detailed analysis (based on their contribution to total fire CDF). Progressively more accurate models are then employed for unscreened locations. The objective of this iterative approach is to focus analysis resources on those locations that contribute most to CDF. For example, in many screening analyses, suppression efforts are initially assumed to fail to prevent fire-induced component damage. This conservative assumption is relaxed only for those locations where a detailed analysis is needed.

When documenting the fire PRA, the results of the analysis are typically presented on a location by location basis, and the key or dominant fire scenarios associated with each plant area are typically described in some detail. This detail usually includes some discussion of plant-specific characteristics driving those scenarios, often including relevant design and operational practices, in addition to key fire protection features credited in the analysis. The importance of providing qualitative as well as quantitative results cannot be overstated, such information ("what can go wrong") is part of the triplet definition of "risk" provided at the beginning of this chapter and is needed when efficiently managing fire risk.

For the purpose of clarity, Figure 5 provides a simplified, once-through picture of the process. In actual practice, the analysis can involve multiple iterations. (In some situations, even early elements might need to be revisited should the results of detailed analysis indicate that

[§§] As one minor difference, NUREG/CR-6850defines two stages of physical fire analysis; namely, a preliminary fire source screening task called "scoping fire modeling" and a follow-up task called "detailed fire modeling." In the ASME/ANS standard, these two tasks have been incorporated into one technical element called "fire scenario selection and analysis".

additional information is needed.) Note also that that the figure does not emphasize the location-by-location nature of a fire PRA; many of the elements are exercised for each plant area.

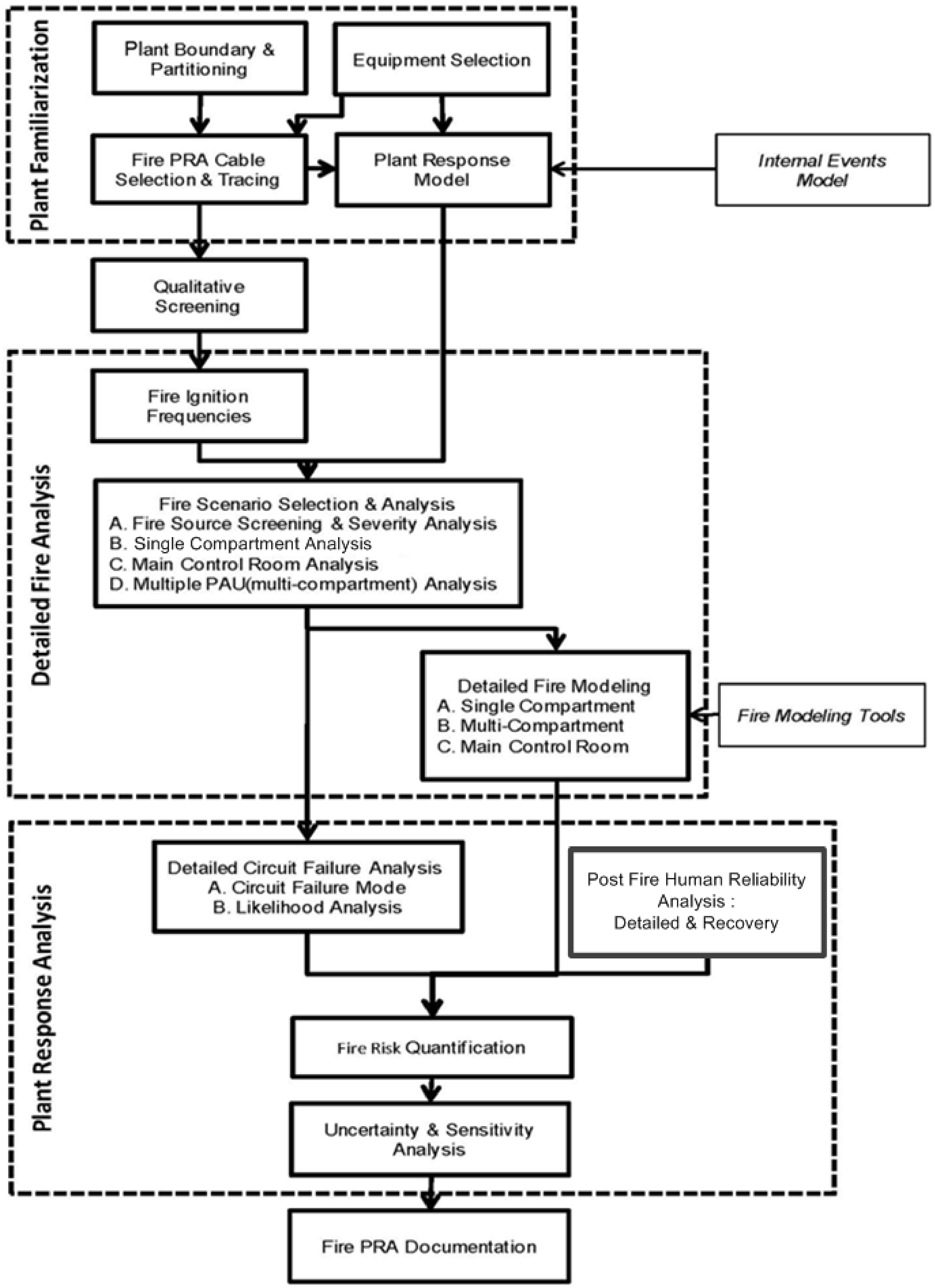

Figure 6- Fire PRA Elements

7. Fire PRA Elements

In this section, we discuss each of the technical elements identified in Figure 5.[***] The discussion provides an overview of current analysis approaches, addresses important sources of uncertainty and ongoing development, and identifies references for further reading.

7.1 Plant Definition and Partitioning

The purpose of the plant definition and partitioning element is to: (a) prescribe the analysis boundaries separating what's included in the analysis (i.e., that portion of the plant addressed in the fire PRA) from what's not included in the analysis, and (b) define how the analyzed plant is to be separated into smaller, non-overlapping spatial units of analysis (spatial partitions). In the language of the ASME/ANS Standard, the first step is referred to as defining the "global analysis boundary" and the second step is referred to as "plant partitioning." The term used in this chapter to identify fire PRA spatial partitions is "plant areas".[†††] Regardless of terminology, the key point is that the fire PRA is typically managed and organized based on where a postulated fire occurs. The intent of the plant partitioning element is to define a set of plant areas, each with specific boundaries (physical or not), that collectively account for all locations within the plant while also ensuring that no location is encompassed within more than one plant area.

Plant areas are ideally bounded by non-combustible barriers, such as fire-rated, or non-fire-rated, barriers. However, plant areas may also be defined on other bases including active features, such as water curtains, and in some cases spatial separation (i.e., extended distances between fire sources or concentrations of in-situ fuels). These distinct plant areas are where heat and products of combustion from a fire will be substantially confined. For example, in typical practice, the main control room will generally be treated as a single distinct plant area whereas areas within the auxiliary building may be segregated into many different plant areas. Note that the potential failure of a credited partitioning feature is addressed in the multi-compartment fire scenario analysis element described later in this section.

The analysis team has considerable flexibility in defining plant areas. For example, it can group several rooms together and treat them as a single plant area, as long as this grouping does not substantially affect the realism of the analysis results. Such grouping can reduce the amount

[***] The ASME/ANS Standard includes a requirement to address Seismic-Fire Interactions that is not shown in Figure 5. Current methods for analyzing these interactions are mainly qualitative and have no direct impact on the fire PRA process or results. Furthermore, since a seismic event is the actual initiating cause, the consequential fire scenarios should, following typical PRA conventions, be included in the seismic portion of the PRA rather than the fire portion. (Of course, care should be taken to ensure the issue does not fall through any gaps.) With recent events in Japan, including seismically-induced fires at the Kashiwazaki and Onagawa plants in Japan, the NRC and the nuclear power industry are considering developments to address the risk of such events quantitatively.

[†††] The terminology surrounding the naming of fire PRA spatial partitions has a somewhat checkered history. Various terms used in fire protection engineering have definitions that were not developed for fire PRA. For example, in fire protection engineering "fire zone" is often defined in the context of coverage areas for a fire detection or suppression system. Similarly, "fire area" has a very specific meaning in the context of the NRC fire protection regulations (see Regulatory Guide 1.189 for related discussions). NUREG/CR-6850 avoids the use of these and other preexisting terms and instead refers the spatial units as "fire compartments," although the fire PRA spatial partitions may or may not be enclosed spaces. The more accurate, albeit fairly technical phrase used in the ASME/ANS Standard is "physical analysis units" or PAUs. Fire PRA spatial units may correspond to a fire zone, a fire area, a compartment, a subset of any of these, or indeed a superset of smaller compartments. For the sake of simplicity, this chapter uses the very generic term "plant area."

of analysis required for risk-insignificant scenarios. In general, the fire PRA process is designed to minimize the level of effort spent on low-risk scenarios and focus analysis attention on risk-significant scenarios. Since the analysis results are not known beforehand, analyst judgment and some iteration are necessary.

From the above discussion, it can be seen that plant partitioning relies heavily upon fire protection engineering expertise, as this is needed to determine the extent to which provided fire protection features will substantially confine heat and products of combustion from potential fires associated with the fire hazards in a given area. Plant partitioning also requires substantial input from other disciplines, as it requires knowledge of the general location and PRA-model relevance of plant systems and equipment (including cables). For example, returning to the above discussion on grouping plant areas, if detailed information for the routing of key cables is unavailable, then a detailed partitioning of a plant area containing those cables may provide little benefit to the analysis.

It is important to recognize that, as in other parts of the fire PRA, plant definition and partitioning is not just a paper exercise. Confirmatory walkdowns are essential to confirm the existence and integrity of the fire protection features and elements credited in defining each plant area.

7.2 Equipment Selection

The equipment selection element defines those pieces of plant equipment and components (other than electrical cables) that will be included in the "plant response model"; that is, the PRA systems model used to perform the plant response analysis. (The development of the plant response model will be discussed later in this chapter.) The selected equipment and components are often referred to as *PRA or fire PRA components*. This element does not start from a blank sheet of paper; rather, the selection of plant equipment for the fire PRA usually begins with the equipment already included in the internal events PRA plant response model. It is common that most, if not all, of the equipment credited for safe shutdown in the internal events PRA is also credited in the fire PRA. The ultimate goal of the fire PRA equipment selection element is to refine the list of credited equipment. The analyst adds new equipment with unique fire induced failure modes such as spurious operation[‡‡‡], not considered in the internal events PRA.

The equipment selection element is performed in close coordination with the development of the fire PRA plant response model. Indeed, it is often the case that choices made in development of the plant response model are what ultimately drive equipment selection decisions. In some cases, these two elements are performed as, in effect, one larger element encompassing the goals of both as described here.

The analysis team performing this element should have collective knowledge of plant systems operation, potential component failure modes, and potential operator responses to different component and instrumentation behaviors. Fire protection engineering inputs include

‡‡‡ Spurious operations are defined as a circuit-fault mode wherein an operational mode of the circuit is initiated (in full or in part) due to failure(s) in one or more components (including the cables) of the circuit; examples are a pump spuriously starting, or the spurious repositioning of a valve.

the above-mentioned fire response procedures and the supporting technical basis for those procedures (including the plant's safe shutdown analysis, developed to ensure compliance with fire protection regulations).

The output of the Equipment Selection Element is a list of equipment (with associated characteristics) to be considered when developing the plant response model, and to be used in identifying key electrical cables, as discussed in the next section.

7.3 Cable Selection and Tracing

The purpose of the cable selection and tracing element is to identify the power, control, and instrumentation cables associated with the components identified in the equipment selection task, plus any other cables whose failure might adversely affect the plant response. Once identified, it is also necessary to determine their physical routing through the plant. This element is usually performed by electrical engineers with extensive plant personnel support.

For reasons previously discussed, risk-significant fire scenarios typically involve damage to electrical cables. The ability of a fire to damage cables decreases rapidly with distance; hence, the more detailed the cable routing information the more realistic the results will be. However, the development of detailed information can be an extremely resource intensive activity depending on the extent and quality of pre-existing information. Clearly, plants that have up-to-date cable routing databases are much better placed to support detailed analyses. Ideally, cables are traced not just to plant areas, but down to the tray/conduit level. This effort has proven to be one of the most costly elements of a fire PRA.

The output of the Cable Selection and Tracing element is not simply a list of cables. This element also establishes, for each cable, a link to the associated plant response model components, to specific component failure modes (e.g., loss of function and/or spurious operation), and to the cable's routing and location. These relationships provide the basis for identifying potential component functional failures at a plant area or raceway level. Inaccurate selection of cables will ripple through the entire PRA and can result in erroneous CDF estimates. It should be noted that many analysts perform this task in conjunction with a detailed circuit analysis (discussed later) in order to avoid unnecessary effort. For example, plant records may "associate" cables with a component whose failure would not, in fact, impact component function (so-called "off-scheme" cables). Identifying such cables early may avoid time consuming cable tracing efforts.

7.4 Qualitative Screening

The qualitative screening element is the first pass made at identifying plant areas that are of such low risk importance that quantitative analysis is not needed. It is not intended to develop CDF estimates for particular plant areas at this stage. It is intended, however, to identify those plant areas where, according to pre-determined criteria (already described above), the fire risk is expected to be very low or nonexistent both compared to other plant areas and in absolute terms. This step can be performed by both system engineers and fire protection engineers once the plant partitioning, equipment selection and cable tracing tasks are completed.

As PRA is an iterative process, if the analyst chooses to modify the list of credited equipment or the cable list at a later stage of analysis, the qualitative screening analysis should also be reviewed to ensure that plant areas initially screened out still satisfy the screening criteria. Plant areas qualitatively screened in this element will be reexamined in a later step for fires that may cause potentially risk-significant damage to equipment located in adjacent plant areas (see the later discussion on multi-compartment fire scenarios).

7.5 Plant Response Model

The plant response model characterizes how combinations of equipment failures and operator errors can lead to core damage. Equipment failure can occur either from causes internal to the component itself (also known as random failures) or from sources external to the component (e.g., fire, earthquake and flooding). As noted above, the plant response model is typically developed by modifying an existing internal events model. This internal events model is typically expressed using event trees and fault trees, as discussed in the PRA Procedures Guide.[42]

As in the case of the internal events model, the fire PRA plant response model supports the analysis of initiating events and ensuing chains of events. Initiating events are conditions that may occur randomly or, in the case of the fire PRA, due to fire-induced component and cable damage. These conditions perturb the balance of plant operation requiring response from automatically or manually activated systems to prevent unsafe conditions. Two examples of initiating events are a loss of offsite power (LOOP) and loss of coolant accident (LOCA). The chains of events ensuing from each initiating event can be represented at different levels of detail. At a high level, as typically represented using event trees (see Figure 2), the events often represent system or functional failures. At a more detailed level, as typically represented using fault trees (see Figure 3), the events can represent failures of specific equipment or operator actions that can contribute to the higher level system or functional failures. These are generally defined in terms of equipment status (failed positions) and erroneous operator actions.

Event trees and fault trees help the analyst to break down potential event progressions into their elemental parts and also to identify a wide range of event sequences. The plant response model specifies many of the important conditions affecting the probability of equipment failures and operator errors, and also provides a mathematical mechanism to estimate CDF based on the equipment failure and operator error probabilities. The event tree and fault tree models can be very complicated, containing thousands of scenarios and a total number of event sequences that easily number in the millions. Dedicated computer programs (e.g., SAPHIRE[43] and CAFTA[44]) have been developed to support the development, execution, review, and documentation of these models.

Since the fire PRA typically starts with an internal events model, the primary concern of the fire PRA analyst is to ensure that the internal events PRA plant response model is appropriately modified to address the special conditions imposed by the fire scenarios analyzed. These conditions include: (1) the failure of plant equipment directly caused by the fire; (2) spurious actuation of equipment due to control circuit failures; (3) new accident sequences that may have

been dismissed in the internal events model as very unlikely to occur given only random failures; (4) new accident sequences due to implementation of plant procedures specific to fire conditions (e.g., alternate shutdown in the event of a main control room fire); (5) instrumentation failures not included in the internal events model; and (6) new or modified plant operator performance related events. (These topics are discussed in some detail later in this chapter.) The information management demands arising during the development and execution of the plant response model can be considerable. Specialized computer programs (e.g., FRANX[45]) have been developed to aid the analysts.

7.6 Fire Frequency Analysis

The fire frequency analysis involves two primary goals. The first is to define representative fire scenarios for the area of interest; that is, to define those fire ignition sources that might lead to risk-relevant fire scenarios. The second is to estimate the frequency of fires (typically events per year) involving each identified ignition source. Depending on the particular objectives of the fire PRA and the potential risk significance of the plant area, the scenarios can be defined in a broad or detailed manner. For example, for plant areas where a conservative analysis is sufficient to demonstrate low risk, the representative fire scenario can be defined as a fire anywhere in the area. For plant areas where a more detailed analysis is needed, multiple fire scenarios defined in terms of the precise location and initial magnitude may need to be specified. The estimation process generally involves the statistical analysis of historical data and the application of engineering judgment.

In many past PRAs, the statistical analysis has been done on an area/location basis[39,40]. However, under current guidance, the analysis is typically done on an ignition source basis. The first approach involves estimating fire frequencies for common plant area types (e.g., main control room, cable spreading room, switchgear room, auxiliary building, turbine building, etc). The second approach involves estimating fire frequencies for common component types (e.g., pumps, electrical panels, transformers) and objects of particular interest to fire analysis (e.g., collections of transient combustibles). Currently, the component-based approach is preferred because it enables the analyst to account for plant-specific differences in the location of ignition sources. NUREG/CR-6850 identifies roughly 40 unique fire ignition source groupings (bins) that reflect both component and source types and, in some cases, the specific fire type to be postulated for a given component type. (For example, pumps may be involved in either oil fires or electrical motor fires, and higher energy electrical switching equipment may be involved in either slower developing "thermal" fires or rapid high energy arcing fault fires.)

The fire frequency for a specified component group (or plant area type) is estimated using conventional statistical modeling methods.[§§§46,47] (See also this Handbook's Chapter

---

[§§§] Consistent with typical PRA modeling practices for many hazards it is generally assumed that fire frequencies are constant over time. One uncertainty regarding fire frequencies is whether or not this is a good assumption. The U.S. nuclear industry has documented fire event data going back to 1965, but the appropriateness of using of older events in a current analysis has been questioned. Prior analyses of fire frequency trends have reached divergent conclusions. One review of fire event data published in 2001[46] showed that the frequencies of reported fires in key U.S. nuclear power plant compartments had not changed dramatically when comparing the periods 1965–1985 and 1986–1994, and thereby supported the use of the Poisson model for U.S. nuclear power

"Reliability.") These methods include direct approaches that assume that all of the industry data for a particular fire type come from the same population and can be pooled, and more sophisticated approaches that assume that there is plant-to-plant variability in fire occurrence rates even for a specified fire type. In the first class of approaches, generic industry frequencies are generated using observations of the form *n* occurrences (events) in time *t,* grouped by the component types defined in the fire PRA. In a Bayesian approach, the generic industry data are used to develop a prior distribution for the fire frequency. This prior distribution is then updated using plant-specific data. Regarding the second class of approaches, a two-stage Bayes method, described by Kazarians and Apostolakis,[48] has been used in developing the frequencies provided in NUREG/CR-6850.

Regarding the data used in the statistical estimation process, reports of fire events at U.S. nuclear power plants beyond those that require reporting to the NRC are provided to Nuclear Electric Insurance Limited (NEIL) by the plant licensees on a voluntary basis. For fire PRA, the event reports are screened to exclude construction fires and other plant fires not relevant to fire CDF (e.g., office building or warehouse fires). The remaining reports were used by EPRI to establish a proprietary fire event database used to develop the fire frequencies reported in the NUREG/CR-6850. Limited fire event information from the EPRI database, modified to ensure plant anonymity, is provided in NUREG/CR-6850. The proprietary database has many shortcomings, one of which being its completeness (since the reporting of many fire events is voluntary). A plant-to-plant variability analysis has been performed to address the impact of apparent differences in reporting. The result was a moderate increase in fire frequencies for all fire ignition sources, but the question of potential underreporting remains an area of uncertainty relative to fire event frequencies. As discussed further below, efforts are currently underway to gather a more complete, comprehensive and consistent set of event data from the entire U.S. commercial nuclear fleet. This work is being performed as a collaborative activity between the NRC and EPRI.

Despite the large uncertainties typically resulting from any of these approaches (due to the small amount of data available for each fire type), it is common practice to use simple point estimates of fire frequencies. The mean values of the uncertainty distributions provided in NUREG/CR-6850 are generally used for this purpose. In recent fire PRAs, these uncertainty distributions are propagated through the risk model only for dominant contributors to estimate the uncertainties in the total fire CDF.

The above discussion focuses on estimating the occurrence rate of fires (regardless of fire magnitude). We note that the severity of a fire event is also an important element of fire PRA. In many early studies, "severity fractions" or "severity factors" were used to address the relative likelihood of "challenging fires," fires that have the potential to cause significant damage in a relatively short amount of time, as a fraction of all the fires included in the fire frequency calculation. One complication of this approach was that, in general, the applied severity factors

---

plant fire occurrences. However, a more recent analysis demonstrated an apparent shift towards lower fire occurrence rates around 1990. The fire PRA community has not reached consensus on the subject, and as discussed in the "Future Directions" section below, an ongoing effort to gather additional fire event data is being performed, in part, to resolve the question.

implicitly took some credit for early fire suppression activities and fire suppression was then independently credited later in the analysis. Care must be taken when analyzing fire detection and suppression (as discussed in the following section) to avoid double-counting their effectiveness. Under current approaches, the event data are pre-screened to eliminate fire events that would not have led to risk-relevant challenges under any foreseeable circumstances. While severity factors remain a part of the current analysis methods, their role in the analysis has changed. Severity factors are now used to reflect aleatory uncertainty regarding the severity of fires that do occur; that is, even for a given fire ignition source, no two fires will develop exactly the same and this is reflected using a severity factor approach. This topic is discussed in the following section.

The component-based frequency approach has not yet reached full maturation. Current practice begins with the generic plant-wide frequency values and then partitions those values within the plant based on the locally calculated population of specific fire ignition source types (e.g., pumps, motors, electrical cabinets, etc.). Hence, it represents something of a hybrid between the earlier location-based approaches (where in this case the location is in effect the entire plant) and a true component-based frequency where a given pump would have a specific frequency value and the plant-wide pump fire frequency would then be based on the number of pumps present at the plant. There are challenges to implementing such an approach and a true component based frequency approach remains a challenge for the future. This is discussed further in the Future Directions section.

7.7 Fire Scenario Selection and Analysis

7.7.1 General Approach

The goal of the fire scenario selection and analysis element is to perform a detailed analysis of the fire growth, propagation and possibility of fire suppression before damage to a specific target set for each applicable fire scenario per plant area. Plant areas that have been previously screened out because the plant area of fire origin does not contain important equipment are reexamined for the potential that fire damage might spread to adjacent, important plant areas. The analysis is based on the evaluation of fire scenarios which are defined based on the general characteristics of the fire being postulated, the potential extent of fire-induced damage, and fire mitigation measures that will be credited for preventing damage

As discussed by Apostolakis et al,[31,32] the assessment models the occurrence of fire damage as the outcome of a "race" between two parallel processes: (1) fire growth and damage, and (2) fire intervention and suppression. To determine the outcome for a particular fire scenario, the analyst needs to address two key questions: "how quickly (if at all) will this fire cause damage if it is allowed to burn without intervention?" and "how quickly will the fire be suppressed?" Since the fire growth and suppression processes are both subject to random variability, this race is a probabilistic process and the probability of equipment damage, the ped,j|i term in Equation (1), can be conceptually represented by the following equation:

$$P_{ed,j|i} = P\left(t_{ed,j|i} < t_{s|i}\right) \tag{6}$$

where:

$t_{ed,j|i}$ = the damage time for target set *j* given fire scenario *i; and*

$t_{s|i}$ = the suppression time for fire scenario *i.*

In practice, the probability of equipment damage is decomposed into two parts as given by:

$$P_{ed,j|i} = SF_i \times P_{ns,j|i} \tag{7}$$

Where

SF_i = severity factor for fire source i; and

$P_{ns,j|i}$ = probability of non-suppression prior to damage to target set j given ignition source i.

The severity factor reflects the fraction of fires involving the fire source that are potentially damaging and the non-suppression factor reflects the probabilistic outcome of the fire damage versus fire suppression race given a potentially damaging fire. Both of these terms are discussed further below.

From a process standpoint, the assessment generally requires:

1. The identification of specific combinations of fire sources and target sets,
2. An analysis of the fire-induced environmental conditions,
3. An assessment of the response of each target set to these conditions, and
4. An assessment of the effectiveness and timeliness of fire mitigation features including detection, suppression and fire barriers.

The fire barrier assessment needs to address barrier effectiveness in preventing fire propagation to adjacent plant areas, as well as in preventing fire damage to protected equipment in the room of fire origin.

In practice, the fire PRA must, at some level, consider any fire that might occur anywhere within the plant. Hence, the analysis may initially consider several hundred to a few thousand potential fire scenarios. The fire scenario selection and analysis element is structured to allow progressive scenario screening to optimize analysis resources by identifying and focusing on those fire scenarios that are the most important CDF contributors. Less important scenarios are analyzed in limited detail and then set aside (screened out) once the conclusion of low CDF importance has been reached. Other fire scenarios are carried forward through the analysis and ultimately quantified[****] as substantive contributors to fire CDF. The decision as to which scenarios to screen out and which to carry forward is typically based on a quantitative screening criteria designed to ensure that the vast majority of the fire-induced CDF is captured by detailed analysis. This process follows the same iterative approach that is commonplace in most fire PRAs. In the end, a typical fire PRA may quantify well over 100 fire scenarios in considerable

[****] In the context of a PRA study, scenarios are "quantified" by estimating their frequencies of occurrence. Ideally, the estimates are expressed in terms of probability distributions for these frequencies.

detail. The next four subsections provide an overview of this analysis process.

7.7.2 Preliminary Fire Source Screening and Severity Factor Analysis

Fire Source Screening is the first task in the Fire PRA framework where fire modeling tools are used to identify and analyze ignition sources that may impact the fire-induced CDF. Screening some of the ignition sources in a plant area may, in effect, reduce the compartment fire frequency previously calculated. This task also introduces the previously discussed severity factor (SF_i). The two main objectives of this task are to screen out fixed ignition sources that do not pose a threat, and to assign severity factors to unscreened fixed ignition sources. This task is performed by fire protection engineers using various fire modeling tools (discussed later) in coordination with methods described in this section with the following material readily available;

- List of components in plant areas,
- Equipment layout drawings, and
- Elevation drawings of rooms and equipment.

The fire protection engineer will also have to establish the characteristics of a credible fire associated with a specific ignition source. The exact nature of the information will depend on the characteristics of the ignition source. The following are examples of such information:

- Quantity of the oil maintained inside rotating machinery,
- Power and voltage of a motor,
- Power of electrical cabinets, and
- Quantity and nature of combustible and flammable materials maintained in an enclosure.

This task is commonly based on plant walkdowns and zone of influence (ZOI) considerations. Regarding the latter, the fire protection engineer specifies a ZOI for the ignition sources in a specific plant area. Each ZOI reflects the potential for an ignition source to cause some level of damage beyond the ignition source itself. Those sources that cannot, by themselves, damage target sets of interest or cause the spread of fire to secondary combustibles are screened out from further analysis.

The analysis should consider the following potential targets:

- The closest component (including cabinets and cables) to the fixed ignition source if no specific knowledge about PRA target locations in the area is currently available;
- Known fire PRA components (targets of interest to the analysis) in the area, if the specific target locations are known; or
- Another intervening combustible material.

These targets can be affected through the following exposure situations:

1. Engulfed in flames,
2. Within fire plume,
3. Within the ceiling jet,
4. Within the smoke layer, or
5. Within the flame irradiation zone.

The type of exposure will depend on the location of the target with respect to the fire. The fire ZOI is defined using fire models to determine the regions where fire conditions will cause target damage or ignition. Conservative fire modeling calculations are then performed to predict the fire conditions near a target to assess if target damage or ignition is possible. The analyst can then be confident that an ignition source can be screened out if no relevant targets receive thermal damage or are ignited. Ignition sources that are part of the fire PRA component list cannot be screened because the loss of the ignition source itself represents loss of at least one PRA target. However, it is rare for scenarios limited to loss of the ignition source to be found risk-significant. Exceptions typically involve electrical cabinets and, in some cases, cable raceways. (It should also be recognized that the internal events PRA model includes the independent component failures, regardless of the failure cause.) For ignition sources that do not screen out, the severity level of the fire needed to cause damage can be established and the corresponding severity factor is estimated.

Fire source burning behavior is commonly characterized by a distribution on peak heat release rate (HRR) reflecting the aleatory uncertainty associated with fire development. Once the peak HRR distribution is developed, higher percentiles of the distribution can be used to estimate the severity factor. For example, if the minimum HRR needed to cause target damage (or ignition, in the case of intervening combustibles) corresponds to the 98th percentile of the distribution, the severity factor for that case is 0.02. This is illustrated in Figure 6, which involves the case of a target within the range of damaging flame radiation. After determining the probability distribution for the peak HRR, the lowest HRR required for damage, $\dot{Q}_{dam}$, is computed. The percentile corresponding to $\dot{Q}_{dam}$ is determined from the probability distribution. The complement of this percentile is the fraction of fires whose peak HRR exceeds the minimum required for damage.

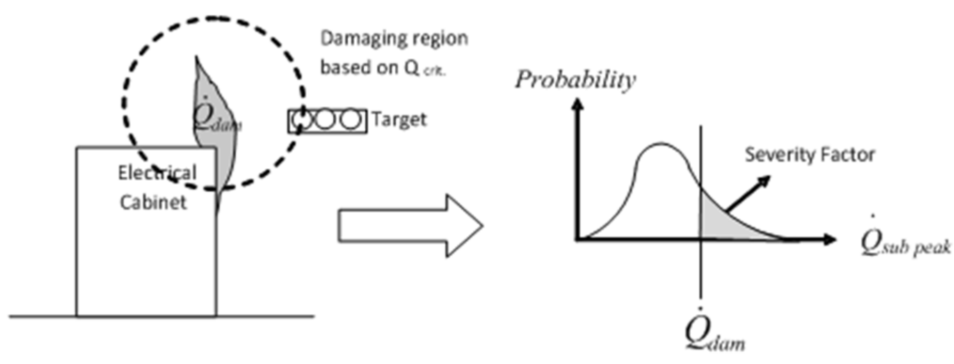

Figure 7- Determination of severity factor using fire models

At this point, the analyst should have a list of unscreened ignition sources with the respective distance to the nearest target and fire condition affecting it (ignition or damage) as well as the corresponding severity factor.

7.7.3 Analysis of Fires Impacting Single Plant Areas

The next step in the Fire Scenario Selection and Analysis element is to perform a more

complete analysis of unscreened plant areas as individual contributors to fire CDF. This task, which is performed by fire protection engineers, involves a detailed analysis of fire scenarios involving unscreened ignition sources in each unscreened plant area that damage target sets located within the same plant area. The majority of fire scenarios analyzed in the fire PRA generally fall into this category.

In early stages of the iterative fire PRA process, the analysis is based on conservative assumptions regarding fire growth and damage. In particular, the fire ignition frequency, plant response (as characterized by the CCDP), and all other relevant parameters are based on the simplifying assumption that any fire in an area would damage all important target sets in that area. In this step, the focus is shifted towards specific damaging fire scenarios impacting a single plant area, and the objective is to estimate their frequencies of occurrence; that is, the frequency of fires leading to the loss of specific target sets within a plant area.

The analysis includes both fixed ignition sources (e.g., fixed plant equipment) and transients. Transients, in turn, include both transient fuel packages such as trash or temporary storage items, and transient activity related ignition sources such as welding and cutting operations.

The initial steps performed in this analysis characterize the plant area and, in practice, are often performed during execution of the Plant Partitioning element. These steps involve:

- Identifying and characterizing the plant area,
- Identifying and characterizing fire detection and suppression features and systems,
- Characterizing fire ignition sources,
- Identifying secondary combustibles, and
- Identifying and characterizing target sets.

Most of this information will be identified during plant walkdowns as well as various equipment selection and cable tracing tasks.

The next steps in this process will largely dictate the resources and time needed to complete the fire PRA. These are:

- Defining the fire scenarios to be analyzed,
- Conducting fire growth and propagation analysis, and
- Conducting fire detection and suppression analysis

In order to define a scenario, the analyst starts with an ignition source, postulates potential growth and propagation to other combustibles, and identifies the target set (or target sets) that may be exposed to the specific fire. The process commonly includes the consideration of an expanding ZOI over time for a single fire ignition source. That is, the ZOI is expanded over time until all relevant targets are included. Another ignition source is then selected and the process repeated until all target set and ignition source combinations are considered. This process should yield all potential fire scenarios that may damage the various target sets identified in the preceding steps. Note that this process represents a “one-to-many” mapping of ignition sources to potential risk scenarios because each target set that is defined has unique implications relative

to plant safe shutdown. That is, each unique target set implies that a unique CCDP value ($P_{CD,k|j,i}$) will be calculated using the plant response model.

The typical analysis approach for detection and suppression is relatively simple. Fixed fire protection systems are credited provided they actuate in time to prevent target damage (based on fire modeling) and are deemed effective against the postulated fire source (based on engineering judgment). Manual fire suppression is typically credited based on historical event data. These topics are discussed in greater detail in the section entitled The Use of Fire Modeling Tools in Fire PRA – Fire Detection and Suppression Analysis.

Depending on the characteristics of the plant area, the analysis may need to treat the potential for dynamic fire growth. Such growth can, over time, cause damage to more and more equipment. This implies more serious plant response challenges and higher CCDP values. On the other hand, the equipment is often spatially separated, and more distance implies longer damage times. Longer damage times, in turn, imply a greater chance that the fire will be put out prior to damage (i.e., a lower non-suppression probability; $Pns,j_{|i}$ – value). A multi-stage approach to assess damage allows for a proper balancing of these two competing effects.

A typical example is illustrated in Figure 7 which shows a stack of cable trays above the fire with fire PRA targets in the first and fourth trays (T1 and T2). A third target tray (T3) is also present but will only be damaged given formation of a damaging hot gas layer.

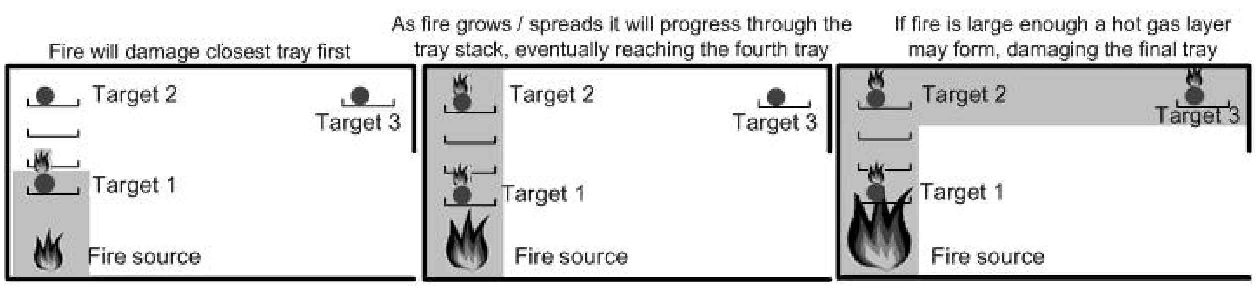

Figure 8- Example case with three PRA targets present that have some degree of spatial separation

The conservative approach to this case is to assume all targets are damaged at the same time but this approach requires that damage be assumed for *all* targets as soon as the *first* target fails (in order to avoid an optimistic result). In this case, all trays would be assumed to fail when the first tray fails. The more complex but also more realistic multi-stage alternative is to represent scenario as a progression of discrete steps accounting for the success or failure of fire suppression efforts within each relevant time frame. Figure 7 illustrates the multi-stage damage state model as applied to the 3-tray example.

It is important that the various damage states be properly balanced in terms of scenario frequency because all arise from the same fire source. One common approach to modeling this type of scenario is through the use of a fire event tree as illustrated in Figure 8. The events in the tree begin with ignition of a damaging fire, and progress through a series of three suppression success/failure events ultimately leading to one of four possible outcomes: namely; no damage to

PRA targets; loss of Tray 1 only; loss of Trays 1 and 2; and loss of all PRA targets (Trays 1, 2 and 3). The branch point conditional probability values based on factors that include, as appropriate, the reliability of automatic detection and suppression systems, timing of each fire damage state, and the performance of manual firefighting. Inherent in the event structure is the condition that the success of an event implies a timely response; that is, the fire must be mitigated before a given damage state is reached. This Handbook's chapter on "Fire Scenarios" provides additional discussion regarding event trees as used in fire protection engineering applications.

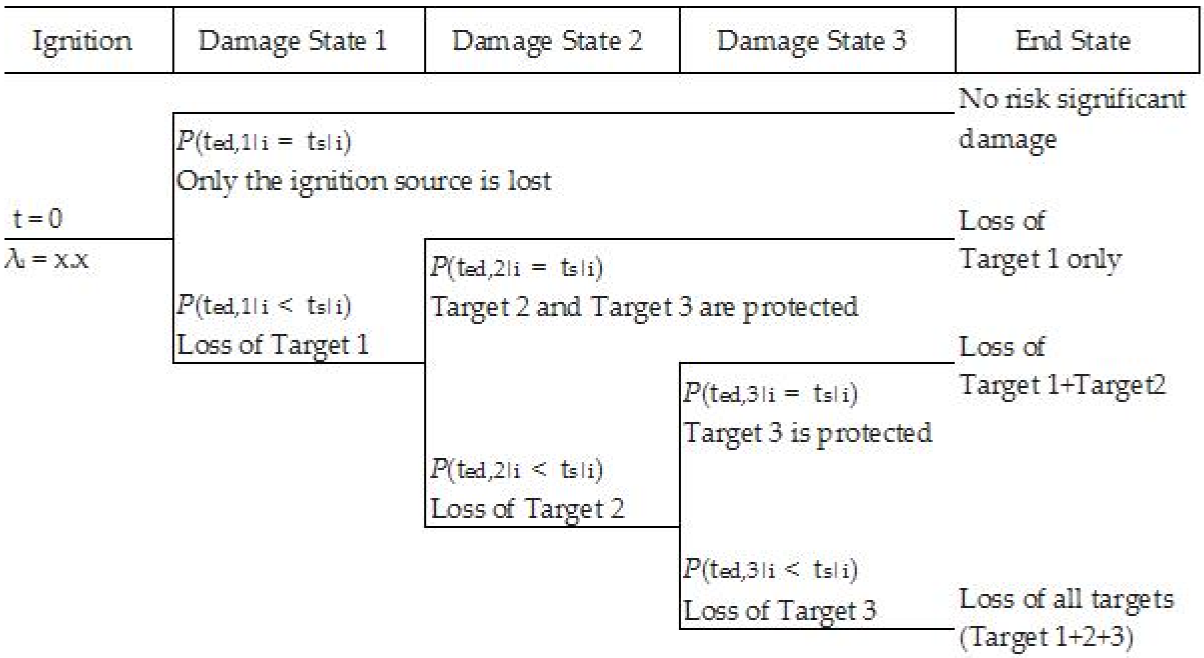

Figure 9- Conceptual event tree representing a scenario progression

A wide range of tools is available for the analyst to conduct fire growth and spread analysis and most of those tools have been described in other Chapters of this Handbook. The tools range from simple empirical equations to computerized, numerical, three-dimensional models. Later in this Chapter, we discuss some of the more common tools used in fire PRA applications.

7.7.4 Analysis of Fires Impacting the Main Control Room

The Main Control Room (MCR) has a unique role in nuclear power plant operations and fire risk, and is typically treated separately from other general plant areas due to their unique fire risk aspects. In MCRs the control and instrumentation circuits of all redundant trains for almost all plant safety systems are present. Furthermore, redundant train controls may be installed within a short distance of one another (e.g., on the main control boards). Therefore, even small fires within control panels may be risk-significant. Additional scrutiny is placed on the MCR also because plant safety depends on the performance of control room operators. A fire adversely affecting the operators' ability to perform needed functions may have severe safety implications. In extreme situations, fire-generated conditions (loss of control functions, high temperature and heat flux, toxic gases, and reduced visibility) could force control room abandonment. All U.S. plants have an alternate shutdown capability, that is, a critical set of independent controls and

instrumentation outside the MCR, to deal with control room abandonment scenarios. Part of the MCR analysis is to analyze the reliability of this capability. On the risk-reducing side, another unique feature of the MCR is that it is continuously occupied with trained operators. This increases the probability that a fire will be promptly detected and addressed.

The MCR analysis task covers all fires that occur within the MCR. This task also covers scenarios involving fires in plant areas other than the MCR that may force MCR abandonment, for example, due to fire-induced loss of a critical set of plant controls and instrumentation. Within the MCR, the target sets consist, for the most part, of control- and instrumentation-related components and wiring within one, adjacent, or nearby electrical control panels and cabinets. As opposed to other plant areas, where targets are usually cables throughout the area, control room targets are cabinets controlling safe shutdown related functions. Cabinets in some MCRs are equipped with smoke detectors, which can reduce the fire detection time and indicate which specific cabinet is on fire. Although MCRs are equipped with smoke detector systems, due to potential risk of spurious activation no fixed suppression is available. Manual suppression is typically the extinguishing method used. Each of these characteristics will impact the risk analysis by influencing the non-suppression or abandonment probabilities.

As compared with the analysis of other plant areas, the unique aspects of the MCR analysis include the consideration of much smaller fire damage footprints (due to the expectation of prompt detection and suppression). For example, a typical scenario in the MCR will assume that damage is limited to a small portion of the main control board rather than assuming loss of an entire electrical cabinet plus external targets. The MCR analysis also considers the impact of fire on operators whereas the analysis of most plant areas considers only equipment and cables. The MCR analysis also involves unique human performance considerations including the decision-making process associated with abandonment. Another unique aspect of the analysis is that, unlike other plant areas, most MCRs are equipped with some form of a smoke purge system. Such systems provide the ability to ramp-up exhaust flow and switch to full fresh-air makeup at a minimum. A well-designed smoke purge capability can delay, or even prevent, forced abandonment. The analysis of such systems calls for more sophisticated models than are typically needed for fire scenarios in general plant areas.

7.7.5 Analysis of Fires Impacting More than One Plant Area

The analysis of multi-compartment fire scenarios covers all fire scenarios involving fire spread from one plant area to another and, therefore, damage in multiple plant areas. Adjacent plant areas are considered systematically and generally in pairs only. That is, based on the occupancy of most nuclear power plant fire areas and the existence of robust fire protection measures, fire spread beyond to a third plant area is generally not considered likely.

Commercial nuclear power plants have numerous interconnected compartments that may be aligned horizontally or vertically. Connections between plant areas include fire doors, stairways, sealed cable and piping penetrations, openings, and gratings. Depending on how plant areas were defined during plant partitioning, plant area interfaces may also include open space (free of combustibles), active fire protection features (such as water curtains or a normally open fire door) and non-rated fire barrier elements. Considering the numerous interconnected compartments and the amount of analysis needed to evaluate fire-generated conditions in multi-

compartment scenarios, a detailed multi-compartment analysis can be resource intensive. Practical analysis, therefore, emphasizes screening multi-compartment combinations before identifying fire scenarios that need to undergo detailed fire modeling.

The analysis often involves combining plant areas of interest into a single area and assessing the potential for fires leading to formation of a damaging hot gas layer in the combined area. This analysis uses standard fire modeling tools, described elsewhere in this Handbook. The analysis also considers paths for direct spread of fire across a barrier element, including the potential that a fire barrier element (e.g., a penetration seal) may randomly fail (e.g., it may have a hole through it due to improperly completed maintenance activities at the time of the fire). The analysis also considers that multi-compartment scenarios will only have a unique risk contribution if new target sets are threatened given fire spread to a second plant area (i.e., as compared to those already threatened in the area of fire origin and encompassed by the single compartment scenarios).

The methods of analysis for multi-compartment scenarios remain relatively unsophisticated, and some areas of uncertainty remain. For example, the random failure probability for a rated fire barrier element is not well characterized. Current practice is to define the plant areas during the plant partitioning task such that risk-significant multi-compartment fire scenarios are unlikely. In theory, regardless of how partitioning decisions are made, the analysis should reach the same conclusions.

7.8 Circuit Analysis

7.8.1 Circuit Failure Mode Analysis

Circuit analysis has become an increasingly complex and important aspect of fire PRAs in recent years. All past fire PRA studies have addressed fire-induced circuit failures that lead to a loss of function and many have included the possibility of fire-induced spurious operation of plant equipment, (although the treatment was limited by current standards. The latter failure mode has become an increasingly important analysis topic over time based on the results of tests and analysis indicating the likelihood of spurious operations, and especially multiple spurious operations, is higher than previously thought.[49-52] Spurious operations can be caused by hot shorts (i.e., electrical faults between cable conductors without a loss of circuit power). Depending on the specifics of the circuit, they can also be caused by other fire-induced cable faults (e.g., single ground shorts or a ground equivalent hot short on an ungrounded power source like station batteries).[44] The likelihood of fire-induced spurious operations and the associated contribution to fire CDF appears to be influenced by a number of plant-specific factors. Although the precise values are uncertain, under any circumstances a modern analysis of these failures is a resource-intensive effort.

Circuit failure mode analysis is performed by electrical engineers. The analysis involves the deterministic failure analysis of important circuits. The purpose of the analysis is to identify those circuits and cables that can adversely affect the credited functionality of essential equipment/components, and to document the equipment responses to the possible cable failure modes induced by fire damage. As noted above, this element is often conducted in close coordination with the Cable Selection and Tracing element, which is aimed at identifying cables.

This element determines the functional impact of potential failures of these cables.

Although this element is performed by electrical engineers, it is useful for fire protection engineers to recognize: (a) the importance of this element to the overall fire PRA results, (b) the potential need to model scenarios involving damage to multiple cables (since multiple cable faults may be needed to induce a spurious operation, and also since multiple spurious operation scenarios might be important to fire CDF), and (c) the particular fire-induced cable faults (shorts to conductors within a single cable, shorts to conductors in another cable, shorts to ground) that may need to be modeled in the analysis.

7.8.2 Circuit Failure Mode Likelihood Analysis

The circuit failure mode likelihood analysis is the probabilistic complement to the deterministic circuit failure mode analysis. That is, this step assigns conditional probability values to specific component failure modes given fire-induced cable failures. Should a fire damage a target cable or set of cables, the likelihood of the above-mentioned cable faults and the durations of these faults are complex functions of many factors, including raceway fill, thermal exposure conditions, fuse size, circuit type, cable construction, and raceway routing. Experiments have been performed to evaluate cable electrical performance for a number of different configurations.[53] The results of these experiments have directly supported deterministically-oriented analyses of fire-induced circuit damage phenomena. In addition, although the tests were not designed to generate a representative random sample, their results provide valuable input to expert elicitation panels tasked with developing estimates for the conditional probability of spurious operations (given fire-induced cable damage) and the duration of these spurious operations. The panels include fire protection engineers (to bring in knowledge regarding both the test conditions and fire modeling approaches for nuclear power plant scenarios), as well as electrical engineers and PRA experts.

The results of an expert panel elicitation for estimating the likelihood of fire-induced spurious operations involving alternating current circuits (AC) are provided in EPRI TR-1006961.[54] An analogous effort aimed at addressing direct current (DC) circuits is ongoing. This latter effort, building on lessons learned from past elicitations, is using guidance on conducting structured expert elicitations originally developed for seismic PRA applications but applicable to other areas as well.[55,56]

7.9 Human Reliability Analysis

In the context of nuclear power plant PRA, Human Reliability Analysis (HRA) is the process used to identify, characterize, and estimate the likelihood of potentially important human errors. These errors can occur prior to the occurrence of an accident (e.g., failure to properly re-align equipment after maintenance), trigger an accident (e.g., inadvertently initiate a reactor shutdown), or during the course of the accident (e.g., fail to appropriately execute a step in plant procedures). HRA has both qualitative and quantitative analysis elements. A general overview is provided in this Handbook's Chapter "Reliability."

In the context of fire PRA, the focus of the HRA element is on errors that may occur as plant operators (including but not limited to the crew of operators in the MCR) respond to fire-initiated scenarios. Human errors causing fires (e.g., inadequately controlled hot work) are built into the fire event data used to estimate fire frequencies, as described earlier in this chapter. Human errors in detecting and suppressing fires are addressed as part of the detailed fire analysis effort. (Such errors affect the time to suppress a fire, denoted by $t_{s|i}$ in Equation (6). Additional discussion on analyzing this term is provided later in this Chapter.)

The specific errors of interest in the HRA element involve operator manual actions taken, in accordance with the operators' procedures and training, to perform needed safety functions (e.g., initiate emergency cooling). These manual actions are typically represented in the plant response model as human failure events (HFEs). HFEs can be provided in event trees (as illustrated by "Manual Reactor Depressurization" branch in Figure 2) or in fault trees (as illustrated by the basic event "Operator Fails to Start/Control RCIC Injection" in Figure 3). They can be errors that are already included in the internal events model (albeit with potentially different probabilities) or they can be introduced into the plant response model due to unique conditions created by the fire scenario (e.g., an inability to complete a desired action due to the fire's location). These conditions are extremely important to the estimation of the HFE probabilities (commonly referred to as Human Error Probabilities or "HEPs"), since human error probabilities depend strongly on the context for the actions being taken. In the case of fire PRA, the analysis team needs to consider such factors as the potential effects of the fire on needed equipment (including such plant support systems as lighting) as well as such physical hazards as heat and toxic gases.

NUREG-1921/EPRI 1023001[57] provides detailed guidance for performing fire HRA. As indicated in this guidance, fire HRA is a multidisciplinary effort. The analysis requires input regarding a wide range of information, including plant procedures[††††], associated operator training, potential fire-induced environmental effects, fundamental psychological and social mechanisms leading to human error, and the "Performance Shaping Factors" (PSFs) affecting these mechanisms. The general steps involve:

1. HFE identification and definition,
2. qualitative analysis of each HFE and,
3. quantitative analysis of each HFE.

The first step identifies those operator actions and associated instrumentation necessary for the successful mitigation of fire scenarios and defines associated HFEs at a level of detail appropriate to support qualitative analysis and quantification.

In the second step, the HRA team develops an understanding as to how each HFE interacts with the overall plant PRA. This understanding should reflect the "as-built, as-operated" response of the operators and plant. Key characteristics include:

[††††] These procedures include Emergency Operating Procedures (EOPs), Annunciator/Alarm Response Procedures (ARPs), and Abnormal Operating Procedures (AOPs), as well as specific fire response procedures.

- Potential fire-induced initiating events,
- Potential accident sequences (particularly functional failures and successes, including preceding operator errors and successes),
- Event timing information,
- Accident-specific procedural guidance,
- Availability of cues and other associated indications that may be needed to identify necessary actions, as well as those that might subsequently enable the operators to detect the need for a correct action that has been omitted or performed incorrectly,
- Preceding operator errors or successes in sequence,
- Criteria defining operator action success, and
- Physical environment information.

The team translates this information into a form useful for the estimation of HEPs. A sound qualitative analysis also allows the HRA to provide feedback to the plant on the factors contributing to the success of an operator action and those contributing to the failure of an operator action.

In the third step, the HRA team develops estimates for the HEPs. This quantification process can be carried out at varying levels of detail, depending on the significance of the HFE. Screening-level analyses using conservative HEP estimates are commonly used for unimportant HFEs, thereby enabling the team to focus attention on those HFEs important to the overall fire CDF.

As with the rest of fire PRA, fire HRA is typically performed in an iterative fashion. Initially simple, conservative analyses are upgraded with more detailed analyses as the team gains a more accurate understanding of the fire CDF and its principal contributors

7.10 Fire Risk Quantification

Risk quantification is where the results of the various elements of fire PRA are assembled to produce the desired risk estimates. This chapter has focused on the fire CDF. Estimates of another risk metric of interest, the large early release frequency (LERF), is developed in a similar fashion to CDF. As its name implies, the LERF metric represents the likelihood of a large, relatively quick release of radionuclides into the environment after a core damage event. A LERF analysis builds on a CDF analysis, but adds models to address the possibility of a breach in the containment structure (see Figure 1) following a core damage accident.

Conceptually, risk quantification is performed by summing the contributions of individual fire scenarios, as shown in Equation (2). In practice, the large number of fire scenarios considered in a typical fire PRA, compounded by the multiplicity of potential plant responses to each fire scenario, calls for the use of specialized PRA software tools to ensure correct treatment of model complexities (e.g., dependencies between different parts of the model) and avoidance of inappropriate conservatism or non-conservatism in the final results. The previously mentioned SAPHIRE[43], CAFTA[44] and FRANX[45] are examples of software packages used in current studies. These programs support the construction and documentation of models, the development of results, and the analysis of results. The last activity includes the identification of

key scenarios, the use of diagnostic ("importance") measures to better understand the contributors to these scenarios, and the performance of sensitivity studies to better understand the impact of different modeling assumptions.

7.11 Uncertainty Analysis

Uncertainty analysis is that element where the key state-of-knowledge uncertainties (often called "epistemic uncertainties") associated with the fire PRA are identified and treated.[‡‡‡‡] Various approaches to uncertainty analysis can be used, depending on the needs of the decision problem addressed by the fire PRA as well as the technical nature of the uncertainties themselves.

In general, the epistemic uncertainties in fire PRA model parameters (e.g., fire frequencies, fire model parameters such as the heat release rate for a specified fire) and in the fire PRA models themselves (e.g., physical models for fire-induced component damage, event tree models for plant safety impacts of fire) are handled as described in this Handbook's chapter "Uncertainty and Safety Factors." Assuming the decision problem (including potential future decision problems) and the role of the fire PRA in addressing this decision problem have already been defined, the uncertainty analysis involves:

- identifying the sources of uncertainty in all aspects of the fire PRA model (including such non-fire protection engineering aspects as the plant response model);
- characterizing the uncertainty in the fire PRA model (both inputs and model structure);
- characterizing the uncertainty in the fire PRA model output; and
- interpreting the results of the uncertainty analysis and presenting this information in a form suitable for use by decision maker(s)

The notion of "characterizing" (as opposed to "quantifying") uncertainties is used to include situations for which a rigorous analysis may not be needed. From a decision support perspective, the key point is not whether the uncertainty analysis results in a mathematically-derived probability distribution for a fire PRA output metric (e.g., the fire-induced CDF), but whether the output uncertainty and its drivers are sufficiently understood to enable selection of the best decision option.

[‡‡‡‡] *As with nuclear power plant PRA in general, fire PRA addresses two types of uncertainty: aleatory and epistemic uncertainty.[10] Aleatory uncertainty, also called "random uncertainty" or "stochastic uncertainty," is that associated with inherent, potentially observable variability in the events and behaviors being modeled. Epistemic uncertainty is associated with limitations in the PRA analyst's state of knowledge, and can involve such things as uncertainties in the true value of an input parameter for a fire model or in the appropriateness of the model itself. Unlike aleatory uncertainty, epistemic uncertainty can be reduced through the collection of additional information (e.g., via experiments).

The fundamental structure of fire PRA is aimed at assessing aleatory uncertainty, as the fire-induced CDF is a measure of aleatory uncertainty (it addresses the likelihood of a random event – the occurrence of a core damage accident due to fire). The uncertainty analysis element discussed in this section deals with the epistemic uncertainty in the fire PRA inputs, models, results, and insights.

NUREG/CR-6850 provides a number of different possible strategies for treating uncertainties. These strategies, some of which can be used in combination, include:

- explicitly quantifying epistemic uncertainties in model parameters using probability distributions, and propagating these probability distributions through the fire PRA model using such techniques as Monte Carlo sampling or Latin hypercube sampling;
- developing multiple models for an issue (e.g., the rate of fire growth for a particular fire), assigning a probability that each model best represents the situation based on engineering judgment, and propagating these probabilities through the fire PRA model;
- identifying a base case as the best estimate model with best estimate data values, performing sensitivity analyses where the models or parameters of interest are varied within a reasonably expected range, and documenting the quantitative effect on the overall results;
- identifying sources of uncertainty that can be (or should be) treated in a single group;
- addressing the uncertainty in only qualitative terms (e.g., describing which scenarios would be affected and providing a qualitative judgment as to the effects); and
- using a quality review process to ensure sufficient accuracy and a reasonable level of completeness (such as a review of the identified cables in a plant area to be sure none have been missed in the PRA model).

The choice of which strategy (or set of strategies) to use may include consideration of the perceived importance of the uncertainty on the overall results of the Fire PRA; the possible effects on future applications or other decision-making activities; and the resources needed and available, including schedule constraints, to execute the strategy.

NUREG/CR-6850 also points out that, given finite analysis resources, not all sources of uncertainties can be treated. The analysis team therefore needs to:

- identify those uncertainties that will not be addressed because they are expected to be unimportant (e.g., they are associated with screened out scenarios) or they cannot be addressed (with reasons noted); and
- identify the strategy (or strategies) to be used to address remaining uncertainties (including which issues that will be treated using sensitivity analysis).

7.12 Fire PRA Documentation

As with any engineering analysis, it is important to document the fire PRA to a level sufficient to enable review of the study by external parties and its use in decision support applications. Specific documentation requirements are provided in the ASME/ANS PRA Standard. These requirements cover such things as the need to document, as appropriate, the methods, data, key factors, modeling assumptions, and results of each of the fire PRA elements discussed earlier in this section

Somewhat unique to PRA (and therefore fire PRA), it is important to recall that the triplet

definition of risk mentioned at the beginning of this chapter includes qualitative as well as quantitative elements. Thus, in general, the results of a PRA need to include descriptions of the scenarios contributing significantly to the overall risk, as well as estimates of the total quantitative risk. These descriptions can be organized along many themes, including the general hazard triggering the scenarios (e.g., fires, floods, hardware failures), the particular initiating event demanding a response from plant safety systems (e.g., unplanned reactor trip, loss of electrical power, loss of coolant), the types of safety functions or systems involved (e.g., low pressure versus high pressure coolant injection systems), or even specific contributing elements (e.g., operator errors).

In the case of fire PRA, a common approach is to present aggregated results for specific plant areas (e.g., the estimated CDF associated with fires in a particular switchgear room, and the uncertainties in the estimated CDF). Depending on the needs of the decision problem addressed by the fire PRA, these area-level results can be further broken down into results for specific scenarios within an area or for specific target sets (e.g., if strategies to protect cable trays from fire damage are being considered).

Another useful presentation of results involves the CCDP for all analyzed plant areas. A given plant area may have many scenarios each with a corresponding CCDP value. However, one value typically generated is the CCDP given loss of all PRA targets in the area. This term, which quantifies the likelihood of core damage assuming that a severe damaging fire occurs, provides a sense of how serious a damaging fire could be. Recognizing that the independent failure probability of typical safety systems can be on the order of 10^{-2} to 10^{-3}, a CCDP in that range indicates that, in addition to the fire damage, an independent failure of an additional safety system would be needed before core damage could occur. This would be an indication that redundancy in safety functions remains even given loss of the entire area. Conversely, a CCDP closer to 1.0 would indicate that a severely damaging fire would be likely to cause core damage by itself. An area with a high CCDP value but a low CDF contribution is also of interest because this indicates that some factor, or factors, have been given substantial risk-reduction credit. Credited factors might typically include assumed fire characteristics, physical separation of PRA targets from fire sources, or fire protections systems and features. Understanding the importance of such factors can be important to decision making.

Armed with these results, decision makers have the information to understand not only the “bottom line” results, but also the reason for the bottom line. This detailed information has proven useful in the development and assessment of potentially effective risk management alternatives.

8. The Use of Fire Modeling Tools in Fire PRA

The preceding section has discussed all of the elements of fire PRA. Most of these elements require at least some input from the fire protection engineer. The two elements requiring the most substantial fire protection engineering involvement lie at the middle of Figure 5: “Fire Scenario Selection and Analysis” and “Detailed Fire Modeling.” Both of these elements require the use of fire modeling tools. These tools are further discussed in this section

8.1 Fire Environment Analysis

Predicting the time to fire-induced failure of PRA targets requires two types of calculations. The first estimates the time-dependent environmental conditions created by the postulated fire in the proximity of the equipment and cables of interest (i.e., the targets). The second addresses the response of the damage target to that environment. This section discusses modeling of the exposure environment; target response modeling is discussed in the following section.

The environmental conditions that a fire model is asked to predict correspond to the target's damage mechanisms. Most commonly, the assessment focuses on temperature and/or heat flux leading to thermal damage. However, other environmental conditions such as heavy smoke may also be of interest, depending on the vulnerability of the target to these conditions. The estimation of environmental conditions requires the treatment of a variety of phenomena as the fire grows in size and severity, including: the spread of fire over (or within) the initiating component (or fuel), the characteristics of the fire plume and ceiling jet, the spread of the fire to noncontiguous components, the development of a hot gas layer, and the propagation of the hot gas layer or fire to adjacent compartments.

The modeling tools used to analyze nuclear power plant fire scenarios are much the same as those described in Sections 2 and 3 of this handbook. Common empirical correlations are employed for the aforementioned phenomena, including even initial estimates of hot gas layer development. The commonly available compartment fire models including both zone models and computational fluid dynamics (CFD) models are also used.

Nuclear plant fire scenarios have a number of characteristics that may not be directly addressed by fire models not explicitly designed to model these scenarios. These characteristics include: a variety of source fire types unique to industrial facilities such as cable trays (see the Cable Tray Fires chapter in this Handbook), electrical cabinets, and pressurized gases; the possibility of propagation through complex fuel arrays such as cable tray stacks; the lack of openings to the external environment; generally complex and often highly congested geometries; the elevated location of many important fire sources and fuels (note that many plume correlations have been based on experiments where the fire is at ground level); and local obstructions such as ventilation ducts, piping, and structural members; and a range of localized fire barriers including raceway fire wraps. By and large, the ability to model such features in detail is limited and often either a conservative approach or an empirical approach are employed in lieu of detailed fire modeling.

Another special aspect of fire risk assessment for nuclear power plants is that fire PRA studies often require that fire modeling be done for many fire scenarios. Within a given plant area, many fire scenarios can be defined depending on the number of fixed and transient combustible sources, the number of unique target sets, the success or failure of fire barriers, and the variety of ventilation conditions, including possible changes (e.g., the opening of a fire door to allow fire-fighting access or the shutdown of a forced ventilation system due to closing of fire dampers). Even a single fire source may be modeled using different assumptions regarding, in particular, peak fire intensity. Given that multiple plant areas may need to be analyzed, and given the potential need to perform calculations to support a quantitative uncertainty analysis, the

number of scenarios to be modeled can be quite large.

A variety of fire-modeling tools, or tool packages, have been developed to address these needs. In U.S. fire PRA studies, including Individual Plant Examinations of External Events (IPEEEs)[22,23,33,34], the most commonly used tools were the package of closed-form empirical correlations drawn from handbooks and provided by the FIVE methodology,[23] the compartment fire zone model COMPBRN IIIe,[58] and the methods described in the FPRAIG.[24] More recent tools include the Fire Dynamics Tools (FDTs),[58] another collection of handbook correlations provided in spreadsheet form and developed to support NRC's risk-informed fire protection-related inspection activities, and the zone models MAGIC and CFAST,[60,61] which has been developed to support nuclear power plant fire PRA. Other general purpose fire models that have been used in fire PRAs include the field model Fire Dynamics Simulator (FDS).[62]

Empirical correlations such as those assembled in FIVE and FDTs are generally used in quick scoping analyses and during ignition source screening analyses. As such, it is preferable that such tools provide conservative estimates but in particular, optimistic results are to be avoided. The zone and field models are typically used to support more detailed analyses where more realistic values are desired.

Information relevant to fire modeling for fire PRA, including data on cable properties (e.g., ignition thresholds, mass burning rates, heat release rates), is contained in several reports.[63-68] Other required parameters (e.g., for cable thermal conductivity, specific heat, density) are typically estimated using data for generic materials. Most zone models follow one of two approaches to modeling the fire source. One approach is to translate all fire sources into an equivalent liquid fuel pool fire (e.g., based on pool size, fuel properties, and fuel quantity). The second approach is to directly specify the fire in terms of a predefined heat release rate versus time. Discussions on appropriate heat release rates for a range of fire sources (e.g., for electrical cabinet fires, based on experimental data reported by Chavez[69] and Chavez and Nowlen[70]) are provided in NUREG/CR-6850 and in a more recent "Nuclear Power Plant Fire Modeling Application Guide"[71] This later report consolidates previously available information and details modeling approaches as they apply to nuclear power plant fire PRA needs.

As discussed earlier in this chapter and in this Handbook's chapter on "Uncertainty and Safety Factors," it is important to treat uncertainties. The predictions of fire models are, of course, subject to significant uncertainties. Applications of fire models that neglect these uncertainties (e.g., applications that neglect the possibility of damage to critical cables because they are a few centimeters above the damage height predicted by a given model, or, conversely, those that neglect the possibility of cable survival when the predicted damage height is just above the cable location) can lead to unrealistic assessments of fire risk. The treatment of uncertainty is needed not only to indicate the degree of confidence in analysis predictions, but can also indicate whether improvements in fire modeling sophistication are likely to change the risk insights for a given scenario. Uncertainty is also an important factor in the assessment of safety margin as called for in the NFPA-805 Standard.

The uncertainties in the predictions of the fire models arise from: (a) "model uncertainties,"

those uncertainties arising from inherent limitations of current understanding of governing phenomena as well as from model simplifications arising when applying fire models to a specific situation, and (b) from a lack of knowledge concerning the actual values of key model parameters.

Regarding uncertainties in fire models, many of the same issues discussed elsewhere in this handbook apply to the analysis of fires in nuclear power plants. For the purposes of this chapter, it is useful to observe that, as discussed earlier in this section, fire PRAs can involve unique concerns and associated complexities (e.g., the spread of a fire over complex fuel beds such as partially filled cable trays, the propagation of fire to secondary combustibles separated from the initial fire source, and the behavior of fires initiated by explosive electrical faults). Available empirical data, for example, those from the 20 ft (6.1 m) separation tests documented by Cline et al.,[72] the German Heissdampfreaktor (HDR) tests (see Nicolette and Yang),[73] the Baseline Validation Tests (see Nowlen[74]), and the recent tests performed as part of the International Collaborative Fire Modeling Project,[75] address only some of these complexities, and do so in a limited fashion. Thus, it is not surprising that there can be considerable uncertainties in the predictions of fire models (including state-of-the-art fire models) when applied in fire PRAs.

Model simplifications, such as assumptions made by software code developers or the code user, are not necessarily from lack of knowledge but rather are routinely made to enable assessments within time and resource constraints. These simplifications can introduce both conservative and non-conservative biases. As examples of conservative simplifications, analyses often ignore (1) the effect of intervening obstacles when calculating heat transfer to a specified target, (2) the heat sink effect of room equipment and the impact of local oxygen starvation on heat release rates, (3) the limiting effect of forced ventilation, and (4) the time required for the fire to reach the "initial" size used to start the fire model simulation. Examples of non-conservative simplifications include the neglect of radiation feedback to the burning fuel in some models and the common assumption that fires in closed metal cabinets will stay confined within these cabinets.

Engineering methods for quantifying fire model uncertainty using integral test data have been developed and applied in a small number of studies.[76,77] A systematic framework for treating model uncertainties, including applications to fire modeling, is provided by Droguett and Mosleh.[78] However, as mentioned in this Handbook's discussions on model uncertainty and in NUREG-1855[11], a quantitative treatment of model uncertainty may not be needed. For example, appropriate efforts to characterize model uncertainty (e.g., through the performance of sensitivity studies to assess the effect of different modeling assumptions) may be sufficient to meet the needs of the fire PRA.

The uncertainty in model parameter values is due to the sparseness of experimental data for some of the parameters (e.g., piloted and non-piloted ignition temperatures for cables) and the uncertainty as to the applicability of the existing data to the situation in the field (i.e., the particular fire scenario being analyzed). Distributions for a number of parameters have been developed (e.g., see Brandyberry and Apostolakis)[79] and used in a number of fire PRA studies. The propagation of these uncertainties through the fire model can be done using readily available tools and methods (e.g., Monte-Carlo or Latin hypercube sampling), as discussed in this

Handbook's Chapter "Uncertainty and Safety Factors."

8.2 Equipment Response Analysis

Given a predicted fire environment for a PRA target, the fire PRA needs to assess the target's response to that environment and determine the timing of equipment failure. Grouped electrical cables present a common cause failure mechanism for multiple plant systems; hence, a key fire PRA concern is the response of electrical cables. However, the responses of other potentially vulnerable equipment (e.g., electromechanical and electronic components in electrical cabinets) are also of interest. Information relevant to the estimation of thermal fragilities of key equipment is provided in a number of reports.[64-66,80-82] Information on the effects of smoke on sensitive equipment is more limited.[83,84] Smoke effects on sensitive equipment are not yet explicitly addressed in detailed fire PRA analyses. (The effects of smoke and fire suppressants on sensitive equipment are implicitly addressed in those screening analyses that assume that any fire within a plant area will damage all equipment within that area.)

Current fire PRA treatments of equipment failure due to heat typically involve very simple thermal models. In the early stages of analysis, it is generally assumed that damage will occur if a representative temperature (e.g., the air temperature surrounding a cable) exceeds a threshold value. Similarly, component damage may also be assumed if the incident heat flux exceeds a critical value. These both represent conservative approaches and a relaxation of this conservatism typically involves the use of simple heat transfer models such as lumped capacitance models or one-dimensional transient heat conduction models. One model recently developed specifically for the analysis of cables is called THIEF.[85] This model is a relatively simple one-dimensional heat transfer model that can take a predicted environmental temperature and estimate the thermal response of cables either in open air or in a conduit. THIEF was calibrated based on an extensive series of small-scale cable exposure tests and validated based on large-scale cable fires. This model incorporates some conservatism because it does not account for the thermal mass effects of grouped cables which will tend to slow down cable heat-up. Instead, the model treats cable bundles as a single target cable. However, the use of even simple thermal response models such as THIEF can substantially reduce analysis conservatism because they account for damage time delays associated with target heating. The longer the predicted time to damage, the more credit can be taken for fire suppression prior to damage.

8.3 Fire Detection and Suppression Analysis

Equation 6 shows that within the context of a fire PRA, the objective of a detection and suppression analysis is to determine the likelihood that a fire will be detected and suppressed before the fire can damage critical equipment. This objective requires an assessment of both the performance of automatic systems and the effectiveness of manual fire-fighting efforts.

Most fire PRA studies performed to date have used a simple detection/suppression model in which automatic systems, if they actuate, are assumed to be immediately effective so long as the fire suppression system is appropriately designed and installed to be effective against the postulated fire. The effectiveness assessment is commonly based on expert judgment and on traditional fire protection design and installations practices (see the guidance provided in the EPRI FPRAIG).[34] The results of calculations for equipment damage times are sometimes

compared with the results of simple fire modeling calculations for fire detector and sprinkler actuation times to determine if automatic systems can be credited. Random failure of the system is also considered. If automatic suppression is unsuccessful (i.e., automatic suppression is not available or, if available, it either fails or is not timely), the likelihood that manual suppression efforts will be effective before equipment damage is then determined. The event data clearly indicate the importance of manual fire fighting because the vast majority of reported fires are suppressed manually rather than by fixed suppression systems.

A weakness with many fire PRA studies that model both automatic and manual suppression is the failure to treat a number of potentially important dependencies between automatic and manual suppression activities. Although the fire water supply system is generally designed such that a single failure of a valve, pump, or piping should not compromise both the primary and backup suppression systems, common cause failure (e.g., due to fire water pumps) cannot be excluded. Most current fire PRA studies do not include a systematic search for such common cause failures. As a second example, if an automatic suppression system were to fail on demand, some delay in manual fire-fighting efforts might occur while personnel attempt to recover the failed automatic system. This dependency has been acknowledged in various methods[9,34] but the actual means for treating this dependency is not well defined and left to the discretion of the analyst. These deficiencies can lead to non-conservative results. Another weakness is that analyses typically neglect delays in fire suppression following fixed-system actuation. However, because the fire growth models used in fire PRA studies do not account for the retarding effects of suppression activities, the risk impact of this neglect is not clear.

Ultimately, manual fire fighting is the final line of defense relative to fire suppression. All U.S. nuclear power plants maintain a fire brigade or fire department. The most common approach to incorporate the fire brigade response in the fire PRA is based on non-suppression curves that are derived from event data. These curves express the probability that a fire will remain unsuppressed as a function of time. An example of such a curve, taken from NUREG/CR-6850, is shown in Figure 10. This particular curve is based on data from all fire events in the EPRI fire events database.[37] NUREG/CR-6850 also provides curves for various specific fire types (e.g., electrical fires, oil fires, etc.) and for certain unique plant areas (e.g., the MCR).

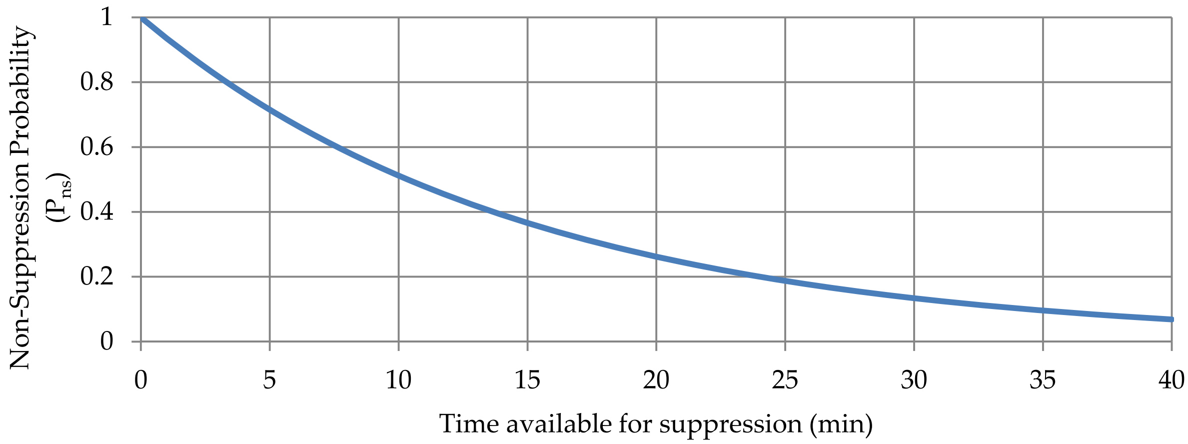

Figure 10- Non-Suppression Probability Curve

The following process is followed to use the non-suppression curves in deriving a probability of non-suppression due to manual means.

1. Using fire models, predict both the time to fire damage and the time to fire detection.
2. Compute the time available for manual fire suppression as the difference between the time to damage and the time to detection.
3. If the computed damage time is less than the computed detection time, set the probability of non-suppression to 1.0 (no suppression credit).
4. Otherwise, use the computed time available on the x-axis of the non-suppression curve to determine the appropriate point on the non-suppression curve and read the corresponding probability of non-suppression from the y-axis.

Note that this simple approach includes an assumption that manual suppression activities begin once the fire has been detected.

This simple approach to modeling manual suppression has some weaknesses. In particular, the non-suppression curves provided in NUREG/CR-6850 neglect scenario and location-specific effects on initial brigade response time. Different plant locations imply different response times for fire fighters to arrive on scene. However, the NUREG/CR-6850 curves are based on generic, industry-wide statistics partly because, for most events, it is not possible to discern the response times. Also, the currently available event records do not provide sufficient information to address plant- and scenario-specific factors such as fire brigade staffing, training, procedures, and equipment available. As a result, the impact of plant-to-plant differences on the fire suppression response cannot currently be reflected in the analysis.

Siu and Apostolakis[86] describe a state-transition methodology that provides a more detailed probabilistic treatment of the detection and suppression process intended to address weaknesses in current practice. This methodology identifies multiple detection/suppression scenarios involving different possible pathways to eventual fire suppression, based on available fire protection equipment for the area of interest. The methodology identifies and treats possible sources of dependencies between elements in these scenarios. Model parameters characterizing

key event times (e.g., the time to suppression) are quantified in a Bayesian framework[10] using generic fire protection system reliability estimates and detection/suppression time data obtained from nuclear power plant fire events. The Bayesian framework provides a direct means for updating model parameters to reflect plant-specific information. (A condensed and somewhat simplified version of this methodology employing data from operational experience has also been developed.[87]) The methodology has been used in a few fire PRA studies (e.g., see Musicki et al.).[12] An alternate methodology was used in a study of the LaSalle plant,[17] but that method (1) does not explicitly identify different detection and suppression scenarios, (2) uses physical models included in FPETOOL[88] to estimate detector and sprinkler actuation times, and (3) uses expert judgment to estimate other characteristic delay times in the fire detection/suppression process.

In NUREG/CR-6850, the state-transition model of Siu and Apostolakis has been restructured in the form of an event tree model, and fire event data have been used to support quantification of the model. As the availability of fire event data has expanded, the data available to support statistical assessments of fire brigade response and suppression times has also expanded. For example, NUREG/CR-6850 provides fire brigade non-suppression curves appropriate for various categories of fires. However, the analysis weaknesses cited above remain. Related activities are described in the Current Activities and Future Directions section.

8.4 Fire Barrier Analysis

As part of determining the immediate environment of equipment potentially affected by a fire, the fire PRA needs to consider the effectiveness of fire barriers. Two general types of fire barriers appear in the analysis. First there are the primary physical structures and features (walls, floors, ceiling and associated penetration seals, doors, hatchways, etc.). Second there are also localized fire barrier systems used to protect plant equipment and cables. The most common localized systems are Electrical Raceway Fire Barrier Systems (ERFBS) including, for example, thermal insulating wraps used to protect cable trays, conduits, or groups of raceways.

The primary fire barriers come into the PRA through the multi-compartment analysis described earlier in this chapter. In the U.S., the most extensive investigation of multi-compartment fires and the effect of inter-compartment barriers was performed by the Risk Methods Integration and Evaluation Program (RMIEP) study.[17] In that study, which was intended to extend the PRA state-of-the-art in a number of areas, the possibility of fire propagation across rated fire barriers between up to three fire areas was treated explicitly. Screening analyses using barrier failure probabilities and assuming the loss of all equipment in all affected fire areas were employed to eliminate unimportant combinations of fire areas. More refined analyses, which distinguished between active barriers (e.g., doors, dampers) and passive barriers (e.g., penetration seals) and employed less conservative barrier failure probabilities (but still assumed the failure of all equipment in all affected areas), were then performed for the remaining combinations of fire areas. The study determined that no combinations passed its CDF screening criterion of 10^{-8} per year, and so multi-area fires were determined to be insignificant contributors to fire risk at the LaSalle plant. One factor in that conclusion is that the plant partitioning analysis was based on physical barriers rather than spatial separation. Hence, in all cases, the inter-compartment barriers were all substantial physical barriers. If an analyst chooses

to credit less robust boundaries (e.g., spatial separation) in defining the PRA plant areas, then multi-area scenarios may prove to be more important.

Regarding the treatment of local fire barriers, these barriers are usually either assumed to be completely reliable for up to their rated fire endurance or are conservatively neglected. It is common, for example, to neglect features such as radiant energy shield and partial height walls. Even when physical models for barrier performance are employed (e.g., COMPBRN IIIe provides a one-dimensional steady-state heat conduction model),[50] these models do not address such behaviors as gross distortion and mechanical failure of the barrier system. Fire tests have shown that such behaviors are strongly affected by installation practices (e.g., the method of sealing joints).[89] Furthermore, the physical properties of the barriers needed to address such complex issues are not readily available.

In typical practice, a fire barrier that is monitored as a part of the plant maintenance program, and that has been demonstrated by testing to provide a specific fire endurance, is credited for providing equipment protection consistent with the fire endurance rating. Again, the most common form of such barriers is raceway wraps. Such wraps can sharply impact the estimated risk. For example, the presence of a one-hour rated raceway barrier systems reduces the likelihood of fire-induced damage to the protected cables substantially. In a typical fire scenario such a barrier can reduce risk estimate by approximately two orders of magnitude (based on manual fire suppression credit given for a fire lasting greater than one hour).

9. Current Activities and Future Directions

Fire PRA currently supports a wide range of activities in the nuclear industry. The most visible activities involve licensee actions to risk-inform their fire protection programs. However, fire PRA support of other activities, both licensee- and regulator-driven, are also important. This section provides a brief overview of both sets of activities, as well as of current research activities aimed at improving the technical basis and performance of fire PRA.

It is important to note that the fire PRA study results (and, more generally, PRA study results) are typically not used as the sole basis for decision making. Other sources of information, including other engineering analyses, are also used to support the decision. In other words, the decision making process is risk informed, rather than risk based. Under this risk-informed approach, the decision maker can make use of information from imperfect or even flawed PRA models, as long as the use of the PRA results and insights improves the decision-making approach for the problem of interest. Siu and Cunningham[90] provide a discussion of challenges of using risk information in fire protection applications and identify a number of lessons from NRC's experience that may be useful to the general fire protection community.

9.1 Risk-Informing Plant Fire Protection Programs

In 2004, the NRC amended Section 48 to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50.48),[15] which governs fire protection programs for operating U.S. nuclear power plants. The amendment added a risk-informed, performance-based option to what had been a completely deterministic rule. In particular, the amended rule allows (with some exceptions specified in the rule) licensees to maintain a fire protection program that complies with NFPA 805 2001 Edition, the National Fire Protection Association standard for a risk-informed, performance-based fire protection program mentioned earlier in this chapter. The purpose of this amendment, as discussed in NRC staff paper SECY-00-0009, "Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking,"[91] was to, among other

things, establish a more reactor-safety-oriented fire protection rule, add appropriate flexibility in some aspects of a licensee's fire protection program, and facilitate the use of alternate approaches that may reduce the need for NRC-approved exemptions from deterministic requirements. The implementation of this rule relies on the licensee's use of a plant-specific fire PRA.

As of mid-2012, as discussed by Harrison et al.,[90] two licensees have received NRC approval for their requests to transition their previously deterministic fire protection programs to the risk-informed, performance-based approach allowed by the amended rule. As part of the approval process, the licensees, the broader regulated industry, and the NRC have learned a number of important lessons, some of which apply to the methods, models, and tools of fire PRA. A number of these fire PRA technology issues were addressed during the approval process and documented in a supplement[20] to NUREG/CR-6850 and other publicly available documents (e.g., the NRC's response to a "frequently asked question" on changes to fire frequencies over time[91]). Other issues are being addressed by EPRI and the NRC's Office of Nuclear Regulatory Research. It is expected that the results of this work will support future updates of NUREG/CR-6850, and thereby the development, review, and approval of the three dozen license amendment requests received or anticipated in the next few years.

9.2 Other Uses of Fire PRA

Fire PRA methods, tools, data, results, and insights have become integral to the treatment of many fire protection and fire-related issues. For example, the results of work on fire-induced cable failures and circuit faults has been used to rank (by risk significance) potentially important factors governing the likelihood of spurious actuations.[92] This risk ranking has allowed the categorization of these factors into sets governing how they will be treated during fire protection inspections or future research. As another example, NRC is using fire PRA results to focus plant fire inspections on the most important fire protection systems and uses a fire PRA– based tool in its Reactor Oversight Program (ROP) to assess the significance of findings from inspections. Using the results of the Significance Determination Process (SDP), the NRC can employ a number of regulatory responses (e.g., fines, heightened levels of regulatory oversight)[95].

More generally, because the risk from fires is a component of total risk, fire PRA can also play an important role in broader risk-informed activities, even those that are not focused on fire. For example, fire PRA is being used in support of analyses supporting the management of changing plant configuration (e.g., to inform decisions as to how long a plant may continue to operate when certain equipment is taken out of service for maintenance). As another example, the NRC's guidance on the use of risk information in proposing plant changes indicates under what conditions a small increase in total CDF could be allowed.[18] If fire is an important contributor to either the baseline (pre-change) total CDF or to the change in total CDF, fire PRA can provide important input to decisions regarding the acceptability of the proposed plant change.

9.3 Uncertainty in Fire PRA and Current Research

As mentioned earlier, the NRC's 1995 PRA policy statement[18] states that:

> "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data ..."

The statement's concern with the limitations of PRA remains important in the case of fire PRA, where research is ongoing to address the still-significant uncertainties in results. (Variations in key analytical assumptions can lead to orders of magnitude variations in estimates of fire-induced CDF and qualitatively different risk insights.)

The uncertainties in fire PRA results are not due to the general analytical approach described at the beginning of this chapter. All current nuclear power plant fire PRAs use this approach or some slight variant on it. Rather, a good deal of the uncertainty is due to weaknesses and gaps in the current treatment of a number of application details and to the assumptions used by analysts when addressing these weaknesses and gaps. Significant uncertainties can arise in the estimation of the likelihood of important fire scenarios (e.g., when addressing the frequency of large, transient-fueled fires or of self-ignited cable fires), identifying initial conditions leading to fire growth (e.g. size of an oil pool fire upon pump system failure), the modeling of fire growth and suppression (challenges include fire propagation through a stack of cable trays and the incorporation of plant-specific factors as fire brigade training and staffing in the treatment of manual fire fighting), the prediction of fire-induced loss of systems (e.g., when quantifying the likelihood of spurious actuations, when addressing the effect of smoke on equipment), and the analysis of plant and operator responses to the fire (e.g., when modeling operator actions during a severe control room fire). Uncertainties also extend to the mode of operation, as no standard approach has been developed to address fire risk at low power and shutdown operations. Research to reduce (or at least better understand) uncertainties in a number of these areas is continuing. Much of this research is being performed as a collaboration between EPRI and the NRC's Office of Nuclear Regulatory Research. The research activities include:

- The development of a comprehensive U.S. fire events database (FEDB)
- The collection of statistics characterizing potential fire sources in U.S. nuclear power plants
- The performance of experimental studies to examine fire effects and behaviors relevant to fire PRA
- The verification and validation (V&V) of currently available fire models
- The development of improved fire PRA models

.Regarding the FEDB, current fire PRAs rely heavily on an EPRI database based largely on voluntary reports.[37] Concerns with the completeness of the reported data have been a limiting factor in the application of event data to fire PRA. The current activity to develop a new, comprehensive FEDB has been planned jointly by EPRI and RES with the actual data gathering activities being led by EPRI. The goal of the effort is to compile a complete list of all fire events at every operating U.S. reactor site occurring between 2000 and 2009. The effort is based on a direct search of plant records. To date, hundreds of thousands of reports have been searched and filtered to identify a few hundred potentially risk-relevant fire events. This effort is nearing completion as of the fall of 2012 with somewhat more than 80% of all reactor sites expected to

be represented in the initial database release.

Once the new FEDB is complete, EPRI and the NRC are planning to use the data to support new estimates of fire event frequencies and of the manual fire suppression curves. The updated fire frequencies are expected to generally follow the fire frequency bins defined in NUREG/CR-6850, although some refinements may be pursued as feasible (e.g., electrical cabinets may be segregated into voltage or functional classes). The effort is also expected to re-examine the question as to whether fire frequencies are changing over time and to address the treatment of historical events in the fire frequency estimation process.

The update of the manual fire suppression curves will be performed to ensure consistency with the fire frequency analysis. If, for example, the fire frequency estimates include many fires of very short duration, then the suppression curves, which are conditioned on the occurrence of a fire, will be quite aggressive (indicating a high likelihood of successful suppression in a short amount of time). If instead very short duration fires are uniformly screened from the fire frequency analysis, then the estimated fire frequencies will be lower but the matching suppression curves would reflect a higher likelihood of longer duration fires.

Regarding plant fire source population statistics, the estimation of component-based fire frequencies requires data characterizing the potential sites for fires (e.g., electrical cabinets, pumps), as well as data for actual fire event occurrences. Many U.S. plants are currently performing fire PRAs and are counting their ignition source populations. EPRI is working with these plants to compile this information.[§§§§]

Regarding experimental studies, work is continuing on a number of projects. These include the Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) project[94], whose data is intended to support the development of improved cable tray fire growth models; the Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire) project,[44] whose data will be used in an expert-elicitation based effort to estimate the likelihood and duration of fire-induced spurious operations in direct current circuits; and an international collaboration with OECD member countries addressing the behavior of high energy arc fault (HEAF) fires associated with electrical switchgear faults. The NRC is also sponsoring research investigating the effectiveness of fire retardant cable coatings to delay fire spread and fire-induced damage, and is planning further research on the behavior of electrical cabinet fires.

Regarding fire model V&V, EPRI and the NRC have jointly addressed a set of fire modeling tools used in current fire PRAs: FIVE-Rev 1,[96] the Fire Dynamics Tools (FDTs),[51] MAGIC,[52] CFAST,[53] and FDS.[54] The V&V effort, which was performed according to ASTM E1355 (“Evaluating the Predictive Capability of Deterministic Fire Models”), selected appropriate fire scenarios for testing the models, established the theoretical basis and assumptions for the models

§§§§ In addition to data collection, the development of a component-based fire frequency approach that goes beyond that provided in NUREG/CR-6850 will require further analysis. This analysis is needed to characterize the relationship between the number of ignition sources and fire frequency, since this relationship may not be linear. (For example, a plant with 50 pumps may not have twice as many pump fires as a plant with just 25 pumps.)

and their implementation, and validated the models over the range of conditions covered by available fire tests. NUREG-1824/EPRI 1911999[96] and NUREG-1834/EPRI TR-1023259[63] provide qualitative and quantitative comparisons of the model predictions versus measurements.[9,63] Currently the NRC, in conjunction with the National Institute of Standards and Technology (NIST), is in the process of updating the V&V effort to include additional data. This effort is intended to expand the validation ranges of the models for some application and to refine current understanding of biases. The results of these efforts are documented during routine updates of the FDS technical reference guide.[98]

Finally, regarding fire PRA models, a range of analytical methods development activities are underway by NRC and various industry groups. These activities are addressing a variety of topics, including manual fire suppression modeling, fixed fire suppression system reliability, multi-compartment analysis, fire barrier analysis, multiple spurious operation likelihood analysis, and the characterization and analysis of transient fires. NRC and industry are working on suitable processes to develop, review, endorse, and promulgate new or revised analysis methods. The vetting process continues to evolve but has already yielded a number of methods improvements and clarifications.

Additional summary-level information can be found in NUREG-1925,[99] which provides an overview of NRC's research efforts in a wide variety of areas (including but not limited to fire safety).

10.Summary

Fire probabilistic risk assessment (PRA) is a systematic quantitative tool for dealing with the complex issues that arise when assessing fire safety at a nuclear power plant. Nuclear power plant fire PRA development efforts date back to the late 1970s. While the supporting methods, tools and data used in the analysis process have been refined substantially, the fundamental analysis framework remains essentially unchanged. The most significant change that has occurred over this time is the manner in which fire risk information is being used. In particular, the NRC's development of a risk-informed option to its previously deterministic fire protection regulations, as part of NRC's broader push to increase its use of PRA and risk in all of its regulatory activities, has increased the attention of industry and regulators on fire PRA.

The increased use of fire PRA in regulatory applications implies a need for higher levels of completeness, realism, accuracy, consistency across the analysis and overall quality. Consistency and realism are goals that continue to challenge fire PRA methods development activities. With fire PRA, the state of knowledge, and the resulting effects on overall realism, associated with various analysis elements remains uneven. A particularly important current challenge involves the treatment of fire-induced spurious equipment operation. This analysis, which involves substantial effort to identify key electrical circuits and analyze the potential effects of fire, holds the potential to significantly alter the insights and quantitative results of a fire PRA. A range of efforts are being undertaken by both the NRC and industry to support the evolutionary process of fire PRA maturation in this and other areas. Development needs and priorities are currently being driven in large part by the desire to increase realism. The overarching goal is to increase analysis fidelity and thereby increase decision makers' confidence

in the risk insights gained.

11. Acknowledgements

The authors gratefully acknowledge the thoughtful comments of Mark Salley, Dave Stroup and David Gennardo.

12.References Cited

1. S. Kaplan and B.J. Garrick, “On the Quantitative Definition of Risk,” *Risk Analysis,* 1, pp. 11–27 (1981).
2. “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” *WASH-1400 (NUREG-75/014),* U.S. Nuclear Regulatory Commission, Washington, DC (1975).
3. “Recommendations Related to Browns Ferry Fire,” *NUREG-0050*, U.S. Nuclear Regulatory Commission, Washington, DC (1976).
4. H. Aulamo, J. Martilla, and H. Reponen, “The Full Stories on Armenia and Beloyarsk,” *Nuclear Engineering International*, 40, 492, pp. 32–33 (1995).
5. E. Pla, “Fire at Vandellos 1: Causes, Consequences and Problems Identified,” in *Proceedings of Fire Protection and Safety at Nuclear Facilities Conference,* Nuclear Engineering International, Barcelona, Spain (1994).
6. S.A. Bohra, “The Narora Fire and Its Continuing Consequences: Backfitting the Indian PHWRs,” in *Proceedings of Fire and Safety 1997: Fire Protection and Prevention in Nuclear Facilities,* London, pp. 219–234 (1997).
7. S.P. Nowlen, M. Kazarians, and F. Wyant, “Risk Methods Insights Gained from Fire Incidents,” *NUREG/CR-6738,* U.S. Nuclear Regulatory Commission, Washington, DC (2001).
8. “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” U.S. Nuclear Regulatory Commission, *Federal Register,* 60, p. 42622 (60 FR 42622) (1995).
9. S. Nowlen, et al., “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” *EPRI TR-1011989* and *NUREG/CR-6850,* Electric Power Research Institute (EPRI), Palo Alto, CA, U.S. Nuclear Regulatory Commission, Washington, DC (2005).
10. G. Apostolakis, “The Concept of Probability in Safety Assessments of Technological Systems,” *Science,* 250, pp. 1359–1364 (1990).
11. “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” *NUREG-1855*, U.S. Nuclear Regulatory Commission, Washington, DC (2009).
12. Z. Musicki et al., “Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1: Analysis of Core Damage Frequency from Internal Fires During Mid-Loop Operations,” *NUREG/CR-6144,* Vol. 3, U.S. Nuclear Regulatory Commission (1994).
13. “Use and Development of Probabilistic Safety Assessment: A CSNI WGRISK Report on the International Situation,” *NEA/CSNI/R(2007)12*, Nuclear Energy Agency (2007).
14. S. Nowlen and T. Olivier, “Methodology for Low Power/Shutdown Fire PRA,” *NUREG/CR-7114,* draft report for comment, U.S. Nuclear Regulatory Commission (2011).
15. “A Review and Evaluation of the Zion Probabilistic Safety Study,” *NUREG/CR-3300*, U.S. Nuclear Regulatory Commission, Washington, DC, (1984).
16. “A Review and Evaluation of the Indian Point Probabilistic Safety Study,” *NUREG/CR-2934*, U.S. Nuclear Regulatory Commission, Washington, DC(1982).
17. J.A. Lambright, D.A. Brosseau, A.C. Payne, Jr., and S.L. Daniel, “Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Internal Fire Analysis,” *NUREG/CR-4832,* Vol. 9, U.S. Nuclear Regulatory Commission, Washington, DC (1993).
18. “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” *NUREG-1150*, U.S. Nuclear Regulatory Commission, Washington, DC (1990).
19. M.P. Bohn and J. A. Lambright, “Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150,” *NUREG/CR-4840,* U.S. Nuclear Regulatory Commission, Washington, DC (1990).
20. “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f),” *Generic Letter 88-20,* Supplement 4, U.S. Nuclear Regulatory Commission, Washington, DC (June 28, 1991).
21. “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, Final Report,” *NUREG-1407,* U.S. Nuclear Regulatory Commission, Washington, DC (1991).
22. “Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program,” *NUREG-1742,* U.S. Nuclear Regulatory Commission, Washington, DC (2002).
23. K. Canavan and J.S. Hyslop, “Fire Probabilistic Risk Assessment Methods Enhancements,” *EPRI TR-1019259 and NUREG/CR-6850 Supplement 1*, Electric Power Research Institute (EPRI), Palo Alto, CA, U.S. Nuclear Regulatory Commission, Washington, DC (2010).
24. J. Lai “Transcript of ACRS Reliability and PRA Subcommittee Meeting July 26, 2012 [Open], Pages 1-292” ACRS Subcommittee Meeting, Agency Documents Access and Management System (ADAMS) ML122260813, U.S. Nuclear Regulatory Commission, Washington, DC (2012).
25. “Fire Risk Analysis, Fire Simulation, Fire Spreading and Impact of Smoke and Heat on Instrumentation Electronics,” *NEA/CSNI/R(99)27*, Nuclear Energy Agency, Paris, France (2000).
26. “Use and Development of Probabilistic Safety Assessment: An Overview of the Situation at the End of 2010,” *NEA/CSNI/R(2012)11*, Nuclear Energy Agency (2012).

27. "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," *IAEA Specific Safety Guide No. SSG-3*, International Atomic Energy Agency, Vienna, Austira (2010).

28. U.S. Code of Federal Regulations, "Fire Protection," *10 CFR 50.48*, August 28, 2007.

29. "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," *RG 1.205, Rev. 1*, U.S. Nuclear Regulatory Commission, Washington, DC, (2009).

30. "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities," *RG 1.200, Rev. 2*, U.S. Nuclear Regulatory Commission, Washington, DC, (2009).

31. "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," *Regulatory Guide 1.174 Rev. 2,* U.S. Nuclear Regulatory Commission, Washington, DC (2011).

32. "Fire Protection for Nuclear Power Plants," *RG 1.189, Rev. 2*, U.S. Nuclear Regulatory Commission, Washington, DC, (2009).

33. Professional Loss Control, Inc., "Fire-Induced Vulnerability Evaluation (FIVE)," *EPRI TR-100370,* Electric Power Research Institute, Palo Alto, CA (1992).

34. W.J. Parkinson et al., "Fire PRA Implementation Guide," *EPRI TR-105928,* Electric Power Research Institute, Palo Alto, CA (1995).

35. "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," *ASME/ANS RA-Sa-2009, Addendum A to RA-S-2008*, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois (2009).

36. "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, National Fire Protection Association," *NFPA 805*, National Fire Protection Association, Quincy, MA (2001).

37. "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Draft Version H, Revision 0, *NEI 07-12*, Nuclear Energy Institute, Washington, DC, (2008).

38. "A Review and Evaluation of the Zion Probabilistic Safety Study," *NUREG/CR-3300*, U.S. Nuclear Regulatory Commission, Washington, DC, (1984).

39. "A Review and Evaluation of the Indian Point Probabilistic Safety Study," *NUREG/CR-2934*, U.S. Nuclear Regulatory Commission, Washington, DC(1982).

40 G. Apostolakis, M. Kazarians, and D. C. Bley, "Methodology for Assessing the Risk from Cable Fires," *Nuclear Safety,* 23, pp. 391–407 (1982).

41. M. Kazarians, N. Siu, and G. Apostolakis, "Fire Risk Analysis for Nuclear Power Plants: Methodological Developments and Applications," *Risk Analysis,* 5, pp. 33–51 (1985).

42. American Nuclear Society and the Institute of Electrical and Electronics Engineers, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessment for Nuclear Power Plants," *NUREG/CR-2300,* U.S. Nuclear Regulatory Commission, Washington, DC (1983).

43. C.L. Smith and S.T. Wood, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8," *NUREG/CR-7039,* U.S. Nuclear Regulatory Commission, Washington, DC (2011).

44. F. Rahn, "Computer Aided Fault Tree Analysis System, Version 5.4", *EPRI 1018460*, Electric Power Research Institute, Palo Alto, California (2009).

45. F. Rahn, "FRANX, Version 4.1", *EPRI 1021231*, Electric Power Research Institute, Palo Alto, California (2010).

46. "Fire Events Database and Generic Ignition Frequency Model for U.S. Nuclear Power Plants", *EPRI 1003111*, Electric Power Research Institute, Palo Alto, California (2001).

47. C.L. Atwood, et al., "Handbook of Parameter Estimation for Probabilistic Risk Assessment," NUREG/CR-6823, U.S. Nuclear Regulatory Commission, Washington, DC (2002).

48. M. Kazarians and G. Apostolakis, "Modeling Rare Events: The Frequencies of Fires in Nuclear Power Plants," in *Proceedings of Workshop on Low Probability/High Consequence Risk Analysis, Society for Risk Analysis,* Arlington, VA (1982).

49. D. Funk and E. Davis, "Characterization of Fire-Induced Circuit Faults," *TR-1003326,* Electric Power Research Institute, Palo Alto, CA (2002).

50. F.J. Wyant and S.P. Nowlen, "Cable Insulation Resistance Measurements Made During Cable Fire Tests," *NUREG/CR-6776,* U.S. Nuclear Regulatory Commission, Washington, DC (2002).

51. J.L. LaChance, S.P. Nowlen, F.J. Wyant, and V.J. Dandini, "Circuit Analysis—Failure Mode and Likelihood Analysis," *NUREG/CR-6834,* U.S. Nuclear Regulatory Commission, Washington, DC (2003).

52. S.P. Nowlen, J.W. Brown, T.J. Olivier, and F.J. Wyant, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," *NUREG/CR-7100*, U.S. Nuclear Regulatory Commission, Washington, DC (2012).

53. G. Taylor, et al., "Electrical Cable Test Results and Analysis During Fire Exposure (ELECTRA-FIRE): A Consolidation of Three Major Fire-Induced Circuit and Cable Failure Experiments Performed Between 2001 and 2011," *NUREG-2128*, draft report for comment, U.S. Nuclear Regulatory Commission, Washington, DC (2012).

54. R.J. Budnitz "Spurious Operation of Electrical Circuits Due to Cable Fires: Results of an Expert Elicitation," *TR-1006961,* Electric Power Research Institute, Palo Alto, CA (2002).

55. R.J. Budnitz, et al., "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," *NUREG/CR-6372*, U.S. Nuclear Regulatory Commission, Washington, DC, 1997.

56. A.M. Kammerer and J.P. Ake, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies," NUREG-2117, U.S. Nuclear Regulatory Commission, Washington, DC (2012).

57. S. Cooper and S. Lewis, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," *EPRI 1023001*, Electric Power Research Institute, Palo Alto, CA, *NUREG-1921*, U.S. Nuclear Regulatory Commission, Washington, DC (2012).

58. V. Ho, S. Chien, and G. Apostolakis, "COMPBRN IIIe: An Interactive Computer Code for Fire Risk Analysis," *EPRI NP-7282,* Electric Power Research Institute, Palo Alto, CA (1991).

59. N. Iqbal and M. Salley, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," *NUREG-1805,* U.S. Nuclear Regulatory Commission, Washington, DC (2004).

60. B. Gautier and C.H. Le Maitre, "User 's Guide for the Softward MAGIC: Version 3.4.2," HT-31/99/007/A, Electricité de France, Paris, France (1998).

61. R. Peacock, W. Jones, P. Reneke, and G. Forney, "User 's Guide for CFAST: An Engineering Tool for Estimating Fire and Smoke Transport," *NIST Special Publication 929,* National Institute for Standards and Technology, Gaithersburg, MD (2000).

62. K. McGrattan et al., "Fire Dynamics Simulator, Technical Reference Guide," *NISTIR-6467,* National Institute for Standards and Technology, Gaithersburg, MD (2002).

63. A. Tewarson, J.L. Lee, and R.F. Pion, "Categorization of Cable Flammability, Part 1: Laboratory Evaluation of Cable Flammability Parameters," *EPRI NP-1200,* Part 1, Electric Power Research Institute, Palo Alto, CA (1979).

64. J.L. Lee, "A Study of Damageability of Electrical Cables in Simulated Fire Environments," *EPRI NP-1767,* Electric Power Research Institute, Palo Alto, CA (1981).

65. L.L. Lukins, "Nuclear Power Plant Electrical Cable Damageability Experiments," *NUREG/CR-2927,* U.S. Nuclear Regulatory Commission, Washington, DC (1982).

66. W.T. Wheelis, "Transient Fire Environment Cable Damageability Test Results: Phase I," *NUREG/CR-4638,* U.S. Nuclear Regulatory Commission, Washington, DC (1986).

67. S.P. Nowlen, "The Impact of Thermal Aging on the Flammability of Electric Cables," *NUREG/CR-5619,* U.S. Nuclear Regulatory Commission, Washington, DC (1991).

68. S.P. Nowlen, "The Impact of Thermal Aging on the Fire Damageability of Electric Cables," *NUREG/CR-5546,* U.S. Nuclear Regulatory Commission, Washington, DC (1991).

69. J.M. Chavez, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effects Tests," *NUREG/CR-4527,* Vol. 1, U.S. Nuclear Regulatory Commission, Washington, DC (1987).

70. J.M. Chavez and S.P. Nowlen, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part II: Cabinet Effects Tests," *NUREG/CR-4527,* Vol. 2, U.S. Nuclear Regulatory Commission, Washington, DC (1988).

71. "Nuclear Power Plant Fire Modeling Application Guide (NPP FIRE MAG) — Draft Report for Comment," *EPRI TR-1023259* , Electric Power Research Institute (EPRI), Palo Alto, CA, *NUREG-1934,* U.S. Nuclear Regulatory Commission, Washington, DC U.S. (2012).

72. D. Cline, W. A. von Riesemann, and J. M. Chavez, "Investigation of Twenty Foot Separation Distance as a Fire Protection Method as Specified in 10CFR50, Appendix R," *NUREG/CR-3192,* U.S. Nuclear Regulatory Commission, Washington, DC (1983).

73. V.F. Nicolette and K.T. Yang, "Fire Modeling of the Heiss Dampf Reaktor Containment," *NUREG/CR-6017,* U.S. Nuclear Regulatory Commission, Washington, DC (1995).

74. S.P. Nowlen, "Enclosure Environment Characterization Testing for the Base Line Validation of Computer Fire Simulation Codes," *NUREG/CR-4681,* U.S. Nuclear Regulatory Commission, Washington, DC (1987).

75. "International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications: Proceedings of 5th Meeting held at National Institute of Standards and Technology, Gaithersburg, MD on May 2–3, 2002," *NUREG/CP-0181,* U.S. Nuclear Regulatory Commission, Washington DC (2002).

76. N. Siu and G. Apostolakis, "Probabilistic Models for Cable Tray Fires," *Reliability Engineering,* 3, pp. 213–227 (1982).

77. N. Siu, D. Karydas, and J. Temple, "Bayesian Assessment of Modeling Uncertainty: Application to Fire Risk Assessment," in *Analysis and Management of Uncertainty: Theory and Application* (B.M. Ayyub, M.M. Gupta, and L.N. Kanal, eds.), North-Holland, New York, pp. 351–361 (1992).

78. E. Droguett and A. Mosleh, "Bayesian Methodology for Model Uncertainty Using Model Performance Data," *Risk Analysis*, *28*, No. 5, pp. 1457-1476 (2008).

79. M. Brandyberry and G. Apostolakis, "Response Surface Approximation of a Fire Risk Analysis Computer Code," in *Proceedings of International Topical Meeting on Probabilistic Reliability and Safety Assessment* (PSA '89), American Nuclear Society, LaGrange Park, IL (1989).

80. J. Wanless, "Investigation of Potential Fire-Related Damage to Safety-Related Equipment in Nuclear Power Plants," *NUREG/CR-4310,* U.S. Nuclear Regulatory Commission, Washington, DC (1985).

81. M.J. Jacobus, "Screening Tests of Representative Nuclear Power Plant Components Exposed to Secondary Fire Environments," *NUREG/CR-4596,* U.S. Nuclear Regulatory Commission, Washington, DC (1986).

82. R.A. Vigil and S.P. Nowlen, "An Assessment of Fire Vulnerability for Aged Electrical Relays," *NUREG/CR-6220,* U.S. Nuclear Regulatory Commission, Washington, DC (1995).

83. T. Tanaka, S.P. Nowlen, and D.J. Anderson, "Circuit Bridging of Components by Smoke," *NUREG/CR-6476,* U.S. Nuclear Regulatory Commission, Washington, DC (1996).

84. R.D. Peacock, T.G. Cleary, P.A. Reneke, and D.C. Murphy, "A Literature Review of the Effects of Smoke from a Fire on Electrical Equipment," *NUREG/CR-7123*, U.S. Nuclear Regulatory Commission, Washington, DC, (2012).

85. K. McGrattan, "Cable Response to Live Fire (CAROLFIRE) Volume 3: Thermally-Induced Electrical Failure (THIEF) Model," NUREG/CR-6931, Vol. 3, U.S. Nuclear Regulatory Commission, Washington, DC (2008).

86. N. Siu and G. Apostolakis, "A Methodology for Analyzing the Detection and Suppression of Fires in Nuclear Power Plants," *Nuclear Science and Engineering,* 94, pp. 213–226 (1986).

87. N. Siu and G. Apostolakis, "Modeling the Detection and Suppression of Fires in Nuclear Power Plants," in *Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Safety Methods and Applications,* San Francisco, pp. 56-1–56-8 (1985).

88. "FPETOOL 3.0," U.S. Department of Commerce, Washington, DC, and National Institute of Standards and Technology, Gaithersburg, MD (1992).

89. G. Taylor and M.H. Salley, "Electric Raceway Fire Barrier Systems in U.S. Nuclear Power Plants," *NUREG-1924*, U.S. Nuclear Regulatory Commission, Washington, DC (2010).

90. N. Siu and M. Cunningham, "Using Risk Information: Lessons Learned from One Agency's Approach," Proceedings of United Engineering Foundation Conference on the Technical Basis for Performance-Based Fire Regulations, San Diego, CA, January 7-11, 2001, pp. 130-140.

91. "Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking," SECY-00-0009, U.S. Nuclear Regulatory Commission, Washington, DC (2000).

92. D. Harrison, A. Klein, H. Barrett, and P. Lain, "Lessons Learned from Risk-Informed, Performance-Based Fire Protection (NFPA 805) Regulatory Reviews," *Proceedings of International Conference on Probabilistic Safety Assessment and Management (PSAM 11/ESREL 2012),* Helsinki, Finland (2012).

93. A.R. Klein, "Closure Of National Fire Protection Association 805 Frequently Asked Question 08-0048 Revised Fire Ignition Frequencies," Staff memorandum, Agency Documents Access and Management System (ADAMS) ML092190457, U.S. Nuclear Regulatory Commission, Washington, DC (2009).

94. K. McGrattan, et al., "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)," draft report for comment, *NUREG/CR-7010*, U.S. Nuclear Regulatory Commission, Washington, DC (2010).

95. "Fire Protection Significance Determination Process," *Inspection Manual,* Chapter 0609, Appendix F, U.S. Nuclear Regulatory Commission, Washington, DC (2005).

96. B. Najafi and F. Joglar-Billoch, "Fire Modeling Guide for Nuclear Power Plant Applications," *TR-1002981,* Electric Power Research Institute (2002).

97. M.H. Salley and R.P. Kassawara, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," *EPRI 1011999*, Electric Power Research Institute, Palo Alto, CA, *NUREG-1824*, U.S. Nuclear Regulatory Commission, Washington, DC (2007).

98. K. McGrattan, S. Hostikka, J. Floyd, and R. McDermott, "Fire Dynamics Simulator (Version 5) Technical Reference Guide," NIST Special Publication 1018-5, National Institute for Standards and Technology, Gaithersburg, MD (2010).

99. "Research Activities 2009," *NUREG-1905*, U.S. Nuclear Regulatory Commission, Washington, DC (2009).

Additional Readings

"Fault Tree Handbook" *NUREG-0492,* United States Nuclear Regulatory Commission, Washington, DC U.S. (1981).

G. Apostolakis, "Some Probabilistic Aspects of Fire Risk Analysis for Nuclear Power Plants," in *Proceedings of the First International Symposium on Fire Safety Science,* Gaithersburg, MD, p. 1039 (1985).

G. Apostolakis, "A Commentary on Model Uncertainty," in *Model Uncertainty: Its Characterization and Quantification* (A. Mosleh, N. Siu, C. Smidts, and C. Lui, eds.), Center for Reliability Engineering, University of Maryland, College Park, MD, pp. 13–22 (1995).

R. Bari and R. Grantom, "Integration and Harmonization of PRA Standards for the Nuclear Industry," in *Proceedings of the International Topical Meeting on Probabilistic Safety Analysis* (PSA '05), American Nuclear Society pp. 503–505 (2005).

H.P. Berg and M. Roewekamp, "Experience from German Reliability Data for Fire Protection Measures in NPPs with Regard to Probabilistic Fire Safety Analyses," in *Proceedings of Fire and Safety 1997: Fire Protection and Prevention in Nuclear Facilities,* London, pp. 63–74 (1997).

D.L. Berry and E.E. Minor, "Nuclear Power Plant Fire Protection, Fire Hazard Analysis," *NUREG/CR-0654,* U.S. Nuclear Regulatory Commission, Washington, DC (1979).

R. Bertrand, F. Bonneval, and P. Lamuth, "Estimation of Fire Frequency from PWR Operating Experience," in *Proceedings of the International Conference on Probabilistic Safety Assessment Methodology and Applications* (PSA '95), Seoul, Korea (1995).

R.J. Budnitz et al., "A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences," *NUREG/CR-6544,* U.S. Nuclear Regulatory Commission, Washington, DC (1998).

L.Y. Cooper and K.D. Steckler, "Methodology for Developing and Implementing Alternative Time-Temperature Curves for Testing the Fire Resistance of Barriers for Nuclear Power Plant Applications," *NUREG-1547,* U.S. Nuclear Regulatory Commission, Washington, DC (1996).

J.C. Helton and D.E. Burmaster (eds.), "Treatment of Aleatory and Epistemic Uncertainty," *Reliability Engineering and System Safety,* 54 (1996).

D. Henneke, E. Kleinsorg, and K. Zee, "Risk-Informed Fire Protection and Fire PRA for Duke Power 's Oconee, Catawba and McGuire Nuclear Plants," *Proceedings of International Topical Meeting on Probabilistic Safety Assessment* (PSA 05), San Francisco (2005).

J. LaChance, S.P. Nowlen, and F. Wyant, "Cable Hot Shorts and Circuit Analysis in Fire Risk Assessment," in *Proceedings of International Topical Meeting on Probabilistic Reliability and Safety Assessment* (PSA '99), American Nuclear Society, LaGrange Park, IL (1999).

J.A. Lambright, S.P. Nowlen, V.F. Nicolette, and M.P. Bohn, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," *NUREG/CR-5088,* U.S. Nuclear Regulatory Commission, Washington, DC (1989).

J. Lambright et al., "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," *NUREG/CR-5580,* Vol. 1, U.S. Nuclear Regulatory Commission, Washington, DC (1992).

P.M. Madden, "Fire Safety Rulemaking Issues Confronting Regulatory Change in the United States," *Transactions of the Structural Mechanics in Reactor Technology Post-Conference Seminar on Fire Safety in Nuclear Power Plants and Installations*, Lyon, France (1997).

F. Mowrer, "Methods of Quantitative Fire Hazard Analysis," *EPRI TR-100443,* Electric Power Research Institute, Palo Alto, CA (1992).

S.P. Nowlen, "A Summary of Nuclear Power Plant Fire Safety Research at Sandia National Laboratories, 1975–1987," *NUREG/CR-5384,* U.S. Nuclear Regulatory Commission, Washington, DC (1989).

S.P. Nowlen and M. Kazarians, "Risk Insights Gained from Fire Incidents," in *Proceedings of International Topical Meeting on Probabilistic Reliability and Safety Assessment* (PSA '99), American Nuclear Society, LaGrange Park, IL (1999).

Proceedings of OECD/NEA Workshop on Fire Risk, NEA/CSNI/R(1999)26, Nuclear Energy Agency, Paris, France (1999).

N. Siu, E. Droguett, and A. Mosleh, "Model Uncertainty in Fire Risk Assessment," in *Proceedings of SPFE Symposium on Risk, Uncertainty, and Reliability in Fire Protection Engineering,* Baltimore (1999).

"Upgrading of Fire Safety in Nuclear Power Plants," *IAEA-TECDOC-1014,* International Atomic Energy Agency, Vienna, Austria (1998).

"Fire Protection and Fire Research Knowledge Management Digest: Printed Brochure" NUREG/BR-0465, Revision 1), U.S. Nuclear Regulatory Commission, Washington, DC (2010).

A Short History of Fire Safety Research Sponsored by the U.S. Nuclear Regulatory Commission, 1975-2008 NUREG/BR-0364, U.S. Nuclear Regulatory Commission, Washington, DC (2009).

The Browns Ferry Nuclear Plant Fire of 1975 and the History of NRC Fire Regulations (NUREG/BR-0361), U.S. Nuclear Regulatory Commission, Washington, DC (2009).

Authors

Dr. Nathan O. Siu is a senior technical adviser for PRA analysis in the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. His major activities involve the development of improved methods for probabilistic risk assessment, including fire risk assessment.

Mr. Nicholas B Melly is a fire protection engineer in the Office of Nuclear Regulatory Research. His major activities involve development of improved methods for fire risk assessment.

Mr. Steven P. Nowlen is a distinguished member of the technical staff at Sandia National Laboratories. His major activities involve fire protection research for nuclear power plants including experimental investigations and the development and application of fire PRA methods, tools, and data.

Dr. Mardy Kazarians is a consultant who has conducted several fire PRAs in the U.S. and abroad. He was a major contributor in the development of the fire PRA methods used in early fire PRAs and still in use today.