A Technical Opinion Paper

Fire PRA Maturity and Realism: A Technical Evaluation[a]

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Abstract

Fire PRA has been characterized as being less mature and less realistic than internal events PRA. Perceptions of immaturity can affect stakeholders' use of fire PRA information. Unrealistic fire PRA results could affect fire-safety related decisions and improperly skew comparisons of risk contributions from different hazards. In this paper, we address the issue of technical maturity through the identification of a number of key indicators and the issue of realism through quantitative and qualitative comparisons of fire PRA results with operational event data. Based on our analysis, we judge that fire PRA is in an intermediate-to-late stage of maturity (albeit less mature than internal events analysis) and that fire PRAs, as performed, may be providing conservative quantitative results. However, our results cannot confidently support estimates of the degree of conservatism. We also observe that the qualitative results of fire PRAs are generally consistent with operational experience. Many of the key issues affecting analysis realism are being addressed by ongoing NRC and industry work; however, we identify a number of additional topics needing work. We note that many PRA practitioners (not just for fire PRA), while recognizing that analysis realism is desirable in principle, see conservatism as an acceptable approach for dealing with uncertainty in practical analyses. Attempts to change such attitudes and beliefs need to be well-targeted and sustained, and success is not guaranteed.

1. Background

1.1 Estimated Risk Importance of Fire

Since the earliest industry-sponsored full-scope probabilistic risk assessments (PRAs[b]) (e.g., the Indian Point Probabilistic Safety Study, reviewed for the U.S. Nuclear Regulatory Commission – NRC – in Ref. 1), and continuing through the NRC's NUREG-1150 [2] and Risk Methods Integration and Evaluation Program (RMIEP) studies [3] and the industry's Individual Plant Examinations of External Events (IPEEEs) [4,5], fire has been shown to be a potentially important risk contributor for U.S. plants.[c] This finding, which is exemplified by the core damage frequency – CDF – mean values provided in Table 1, is not unique to the U.S.; international studies also recognize the risk importance of fire [6,7].

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[a] *This white paper is an extended version of a similarly titled paper presented at the American Nuclear Society PSA 2015 conference (April 26-30, 2015). It has been modified from a 2016 version (available from the NRC's Agencywide Documents Access and Management System – ADAMS – via Accession Number ML16022A266) to correct a figure citation.*

[b] *Appendix A provides a list of the acronyms and abbreviations used in this paper.*

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

Table 1. CDF mean values from early U.S. fire PRAs

Study[a]	Plant(s)	Mean Fire CDF (/ry)	% Fire Contribution to Total CDF[b]
Indian Point (1982)[c]	Indian Point 2 Indian Point 3	2.0E-4 9.9E-5	38 53
Millstone (1983)[c]	Millstone 3	5.0E-6	8
Oconee (1984)[c]	Oconee 3	1.0E-5	4
Three Mile Island (1987)[c]	Three Mile Island 1	1.0E-4	18
NUREG-1150 (1990)	Surry 1 Peach Bottom 2	1.1E-5 2.0E-5	14[d] 72[d]
RMIEP (1992)	LaSalle 2	3.2E-5	32
IPEEE (mid-late 1990s)	99 units	3.7E-5[e]	26[e,f]

[a]The 1981 industry-sponsored Zion Probabilistic Safety Study, a companion to the Indian Point study, performed a more limited assessment of fire risk [8]. Other early, limited scope analyses include the treatment of large electrical fires in the NRC's Reactor Safety Study (WASH-1400) [9] (this assessment assumed the same extent and timing of equipment loss as observed in the 1975 Browns Ferry fire), and the General Atomic Co. analysis of cable spreading room fires [10], performed in support of a PRA for a high-temperature gas-cooled reactor.
[b]Computed as the ratio mean fire (at-power) CDF/mean total (all hazard, at-power) CDF.
[c]Fire CDF estimates developed by licensees are documented in NRC staff and contractor reviews.
[d]Total CDF computed using seismic CDF based on Electric Power Research Institute (EPRI) seismic hazard curves.
[e]Population average for plants performing fire PRAs.
[f]Computed for plants performing seismic PRAs (not seismic margins assessments).

In 2004, the NRC modified its fire protection rule, 10 CFR 50.48 [11] to provide licensees with a voluntary, risk-informed option for meeting the NRC's fire protection requirements. This rule change incorporates by reference (with limited exceptions, modifications, and supplementation) the National Fire Protection Association (NFPA) Standard 805 "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (commonly referred to as "NFPA 805") [12].

Several licensees have submitted Licensing Amendment Requests (LARs) to take advantage of this change. The LAR submittals largely employ the fire PRA guidance documented in the report EPRI 1011989/NUREG/CR-6850 (often referred to as "NUREG/CR-6850") developed jointly by EPRI and the NRC's Office of Nuclear Regulatory Research (RES) [13], and a supplement to that report capturing lessons learned from pilot submittals [14]. The LAR submittals indicate that the estimated fire CDFs remain significant (see Table 2).[d]

The last line in Table 2 is noteworthy, particularly in comparison with the last line in Table 1. We will return to this point in Section 4.1.4 of this paper.

[d] *Note that a number of the LAR CDFs reflect credit for planned changes aimed at significantly reducing fire CDF.*

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

Sample size	34 units
Submittal dates	2011-2014
Average reported fire CDF	4.6E-5/ry
Minimum reported fire CDF	1.1E-5/ry
Maximum reported fire CDF	8.7E-5/ry
% fire contribution to total CDF	35-98%

[a]The CDFs are current as of March 23, 2015. Some of the CDFs are the results of calculations performed following the original LAR submission (e.g., in response to NRC questions).

1.2 The Problem

Although, as shown in Table 1, fire was analysed in many of the earliest PRAs, fire PRA has been characterized as being less mature and less realistic than internal events PRA.[e] For example, in its 1998 review of the NRC's safety research program [16], the NRC's Advisory Committee on Reactor Safeguards (ACRS) stated that:

> *"… [fire PRA] methods have not been as actively developed, debated, and refined as have the methods used for usual risk assessment of power operations… If these [fire-protection related] regulatory activities are to have the benefit of risk information, reliable fire risk assessment methods must be available."*

More recently, the discussion has become more pointed. In 2008, an industry report addressing early lessons learned from efforts to support the implementation of NFPA 805 [17] stated

> *"The current fire PRAs being developed consistent with NUREG/CR-6850 and the additional clarifications provided in reponse [sic] to industry [questions] have more systematic modeling conservatism than the more mature internal events PRA evaluations."*

In 2010, the Nuclear Energy Institute (NEI) provided a report supporting its view that fire PRAs performed using EPRI 1011989/NUREG/CR-6850 and its supplement are conservative [18]. The report states:

> *"…it has become evident to the industry practitioners that:*
>
> - *The manner in which fires are characterized does not conform with operating experience;*

[e] *As defined in the PRA Glossary (NUREG-2122 [15]), internal events analysis deals with scenarios involving the "failure of equipment as a result of either an internal random cause or a human event which perturbs the steady-state operation of the plant and could lead to an undesired plant condition." Internal events are treated as a separate hazard group from other internal hazards (e.g., internal fires, internal floods) and external hazards (e.g., earthquakes, high winds, external floods).*

- *The level of quantified risk is overstated, as compared to operating experience; and*
- *There is an unevenness in the level of conservatism in the results that can mask key risk insights and result in inappropriate decision-making."*

In 2011, an ACRS report on the current state of NFPA 805 implementation [19] first cautioned about generalized labels, stating:

> *"…contentions that the basic fire analysis concepts and methods are 'immature' are an over-simplification. The guidance in NUREG/CR-6850 consolidates advancements in fire analysis methods and knowledge that have evolved over the last 25 years. In some cases, those advancements are rather substantial."*

but then followed with:

> *"…although the methods that are summarized in NUREG/CR-6850 represent the consensus state-of-the-practice, the actual experience base for practical application of those methods was very 'immature' when … the pilot studies began…There is rather compelling evidence that the results from the seven 'mature' fire PRAs that were summarized for our subcommittee investigations are numerically conservative, compared with documented industry operating experience."*

In 2013, in a letter to the NRC [20], NEI identified a number of impediments to the advancement of risk-informed regulation, specifically referring to the "chilling effect" of issues arising in the implementation of NFPA 805, including:

> *"…untested PRA fire methods laced with conservatisms in the required fire-risk analyses… [resulting in] fire PRAs … not consistent with operating experience."*

Gallucci [21] documents and analyses a number of statements (many from public meeting transcripts) regarding the issue of fire PRA maturity and conservatism. Some of these statements assert (as above) that fire PRA methods are relatively immature and produce conservative results, while others provide a moderating or even contrary point of view.

The two related (but non-identical, as discussed in the following section) issues of fire PRA maturity and realism are important practical matters. PRA results and insights are being increasingly used in regulatory applications. These applications range from plant-specific (e.g., the management of plant maintenance activities, the approval of changes to a plant's licensing basis, the assessment of the significance of inspection findings) to industry-generic (e.g., the assessment of potential safety issues affecting more than one plant, the determination as to whether new regulatory requirements should be imposed on the industry). Depending on the particular application, a variety of PRA outputs, including importance measures, accident frequencies (both CDF and large early release frequency – LERF), changes in accident frequencies, and relative contributions to risk may be called for. Clearly, if the analysis of a risk-significant hazard (or hazard group) is unrealistic, the PRA could be providing faulty information to the decision making process. Moreover, an unrealistic analysis could skew comparisons of risk contributions from different hazards, thereby distorting our understanding of risk and degrading one of the major benefits of PRA, which is to help focus attention on areas of "true safety significance" [22]. Even further, if the PRA analysis of a hazard is viewed as immature (or less mature than analyses of other important hazards), stakeholders might be tempted to overly discount even useful information from the PRA in lieu

of evidence from other sources (e.g., statistical estimates based on international operating experience, worst case analyses) that may have their own, if less thoroughly examined weaknesses.

1.3 Purpose

In this paper, we present our views on the maturity and realism of current fire PRAs. The intent of this paper, which updates our analyses in earlier papers [23,24] is not to further inflame passions in a still-heated discussion, but to document our understanding of the current situation. Recognizing that fire PRA concerns continue to shape industry and NRC discussions aimed at improving risk-informed regulation [20,25] and that fire PRA is likely to play a major role in NRC's ongoing Level 3 PRA project [26-28], we believe that our effort is timely.

2. Maturity and Realism in a PRA Context

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

3. On the Maturity of Fire PRA

Judging the maturity of a technical field is a subjective matter, being dependent on the judgment of the assessor. Different authors have identified a number of characteristics they consider to be indicators of maturity. Stetkar and his co-authors distinguish between the maturity of the fire PRA technology (which dictates what level of analysis is possible) from the maturity of the application of that technology (which indicates what is happening in actual practice) [19]. They also tie the notion of maturity to the number of experienced analysts performing fire PRAs. Budnitz provides similar indicators in a discussion of the state of seismic PRA [32], referring to the number of practitioners (or groups of practitioners), the degree

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[f] *In his 2003 speech "Realism and Conservatism," then-NRC Chairman Diaz defines the term "realistic" as "being anchored in the real world of physics, technology and experience" [29]. In a PRA context, because a) the PRA needs to deal with rare (and hopefully unobserved) events, and b) the purpose of the PRA is to support decision making, we think it appropriate to tie the notion of "realism" to the needs of decision making.*

[g] *For example, NUREG-1855 [30] advocates the use of consensus models coupled with sensitivity analyses to address key model uncertainties.*

[h] *Interestingly, the Lewis Commission's 1978 review of the seminal 1975 Reactor Safety Study (WASH-1400) also raised a concern with "a pervasive regulatory influence in the choice of uncertain parameters" [31].*

of practice, and the state of technical development of the field (including the availability of detailed guidance for new practitioners). Budnitz particularly emphasizes the use of the technology in support of practical decision making as an important indicator of maturity. This emphasis is echoed in a Technical Opinion Paper issued by the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations (NEA/CSNI) [33]. Finally, in an exposition on the state of structural safety engineering, Cornell describes characteristic situations associated with the different stages of development of a technical field based on his observations from a number of fields (including geotechnical engineering, structural dynamics, and finite element analysis) [34]. Table 3 provides our synthesis of Cornell's discussion, grouping his situations into one of three categories of indicators involving the field's practitioners, research agenda, and applications.

Table 3. Indicators of stages of technical maturity (adapted from Cornell [34])

	Developmental Stage		
	Early (Infancy, Emerging)	**Intermediate** (Adolescent, Developing)	**Late** (Mature, Stable)
Practitioners	• Small research community • Small number of practitioners • Strong personality influences, competing schools of thought	• Larger number of practitioners • Larger number of experienced researchers	• Many well-trained and experienced practitioners • Recognize limits of applicability of methods • Can adapt methods to new situations • Can work with researchers to identify important issues
Research Agenda	• Driven by perceived needs • Problem selection affected by personal choice (e.g., due to ease of formulation or solution)	• New practice-driven research problems • Some consensus positions for some broadly defined problem areas • Some unproductive research lines abandoned • Incomplete coverage of topics	• Most research driven by needs of practice • More abstract research addresses needs clearly identifiable by all concerned
Applications	• Local applications (addressing small parts of larger problems) • No broader framework	• Fast growth • Developing vocabulary • Optimistic views on new methods; limitations not well understood	• Vocabulary has evolved • General framework exists • Little "selling" of area

Applying the preceding ideas to our experience in developing and applying fire PRA methods, models, tools, and guidance, it appears to us that nuclear power plant fire PRA is: a) in an intermediate-to-late stage of development (i.e., well past the early stage), and b) less developed than internal events PRA.

A key factor in the first part of our assessment is the acceptance of fire PRA results in supporting major decisions, starting with the Commission's 1985 decision to allow continued operation of the Indian Point Plants [35] as informed by findings and recommendations of the NRC's Atomic Safety and Licensing Board (ASLB) [36],[i] continuing with plant changes identified in the IPEEE program[j] [4,5] and more recently with staff approvals of licensee-requested fire protection program transitions as per NFPA 805.[k] These show that the technology is being used in practical applications. Further, it appears that the field has many of the late-stage characteristics identified by Cornell [34]. In particular, a general analysis framework, vocabulary, guidance, and standards all exist [38]; there is recognition of the limits of applicability of existing methods (as indicated by discussions on various areas of weakness, e.g., [18,19]) and of feasible changes for new situations (see, for example, discussions on the adaptation of at-power methods, models, and data for low power and shutdown analyses [39]); and ongoing research is being driven by the needs of practice (as discussed in Section 4.3 of this paper).

Key factors in the second part of our assessment are the relatively small number of fire PRA practitioners (as compared with internal events), the current controversy with a number of the consensus models and data as provided by EPRI 1011989/NUREG/CR-6850 and related guidance, and the lack of consensus regarding the realism of the overall fire PRA results. We recognize that, as pointed out by Stetkar et al. [19], the ongoing licensee and staff activities related to NFPA 805 will increase the fire PRA experience base, and should, over time, reduce the maturity gap with internal events.

It is important to note that, in our view, the mere presence of significant uncertainties does not indicate that a field is immature. There are considerable uncertainties in the treatment of various internal events topics (e.g., reactor coolant pump seal failures, common cause failures, operator errors of commission), yet the internal events analyses are deemed sufficiently mature to support decision making. As indicated in Table 3, the influence of uncertainties on our maturity assessment is primarily expressed through the indicators associated with the research agenda of a field.

Of course our assessment is subjective; others can review the available information and reach a different conclusion. Given that the issue of maturity tends to be self-resolving as long as there are practical application needs and therefore both resources and desire to address weaknesses, perhaps such differences of opinion shouldn't matter very much. However, should discussion be desired, or, more practically, should we wish to accelerate the maturation process, we suggest that a structured consideration of indicators such as those we've identified above can be useful. We note that these indicators suggest several possible actions one could take to increase the maturity of a field – research and development aimed at improving the analytical technology is only one such action. The indicators also support the

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[i] *In the Indian Point PRA, which played a major role in the ASLB hearing, fire was shown to be a major contributor to CDF (see Table 1). The ASLB, in its remarks on PRA areas needing modelling improvement, mentioned fire as one of a number of areas needing improvement (the others were the treatment of operator diagnosis of accidents in progress, direct current (DC) power supply failures, common mode failures due to plant maintenance, seismic hazard, and hurricane and tornado hazard), but did not place special emphasis on this point [36].*

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

[k] *As of March 2015, eleven NFPA 805 LARs have been approved by the staff and 16 are in various stages of review [37].*

point made by Stetkar et al. [19] and others (see, for example, the quotes provided by Gallucci [21]), that substantial changes in fire PRA maturity are likely to take many years; there is no "quick fix."

4. On the Realism of Fire PRA

Fire PRA, as with PRA in general, is aimed at identifying risk-significant scenarios and quantifying their likelihoods and consequences. In principle, it can address scenarios with a wide range of consequences (e.g., various states of plant damage). In practice, the analytical resources of U.S. fire PRAs are typically focused on scenarios leading to core damage and (in recent times) large, early release. To accomplish this, the analysis, as originally formulated [40,41] and currently practiced [13,38], is iterative. Potentially important scenarios are identified, conservatively assessed, and passed on to more detailed analysis stages if they meet certain screening criteria (see Figure 1). The intent is that the overall results of the analysis be sufficiently realistic for the purposes of the study; there is no guarantee that the analyses of non-contributing scenarios, some of which may be important contributors to intermediate end states (e.g., loss of specified safety functions but not core damage), are realistic.

The strong tie of the analysis results to the specific purpose of the analysis complicates our assessment of realism. In this section, we look at this topic from a number of angles: the summary and detailed outputs of past and recent fire PRAs, the technology of fire PRA, the fire PRA applications environment, and the infrastructure supporting fire PRA applications.

4.1 Fire CDF Estimates

One natural approach to assess the realism of fire PRA is to compare its summary output measures (notably, fire CDF) against appropriate empirical benchmarks, e.g., statistical estimates of fire CDF derived from operational experience. However, although there have been a number of "close calls" worldwide, including, but not limited to, the 1975 Browns Ferry fire, there have been no fire-induced core damage events (at least of the type of interest to U.S. fire PRAs[1]) [42]. Thus, a meaningful statistically-based comparison is not straightforward.

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

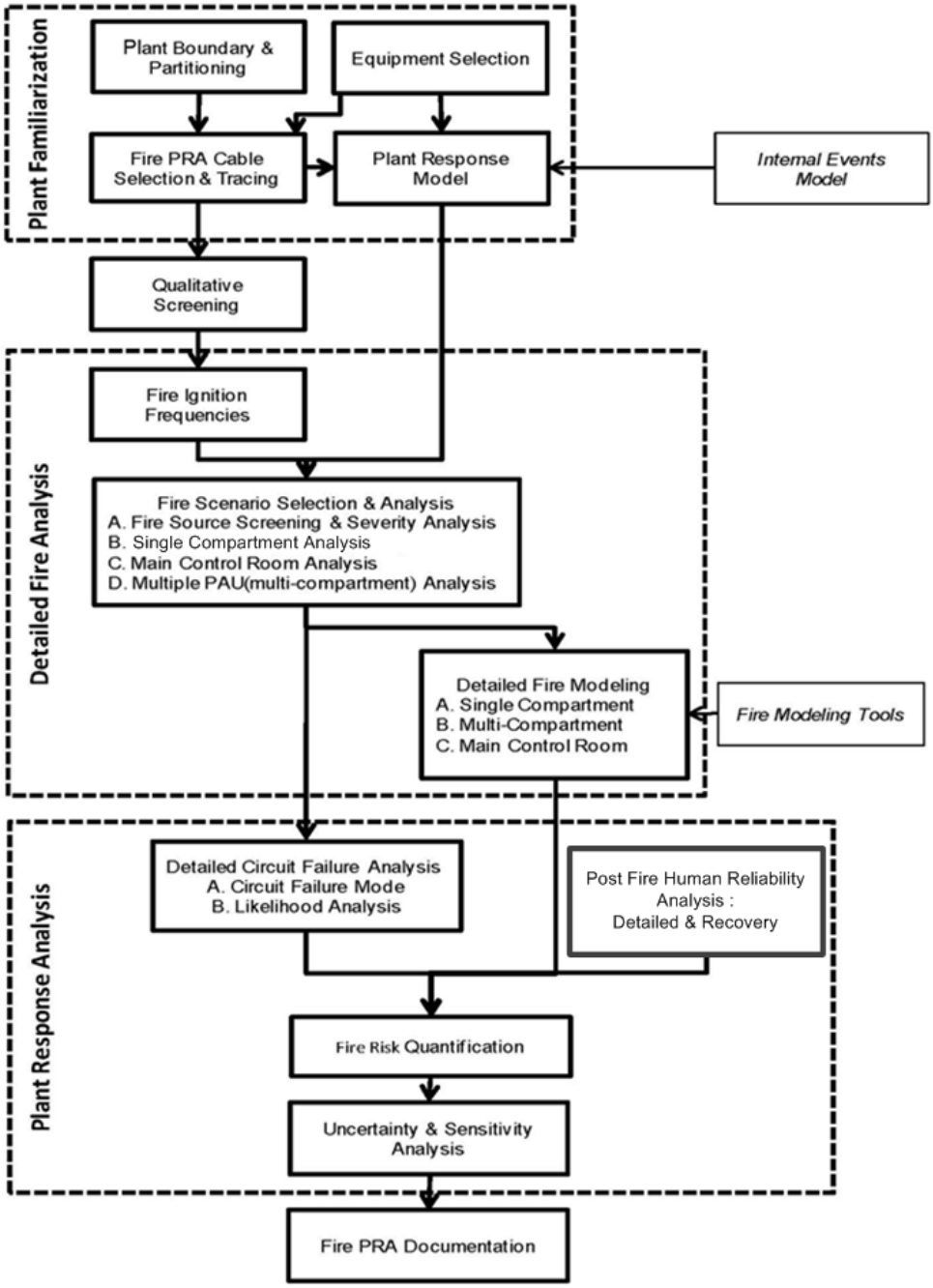

Figure 1. Fire PRA Elements [38]

4.1.1 Statistical Estimates of CDF and the Challenge of Exchangeability

The most direct way to estimate an industry average ("global") fire CDF from operational experience is to divide the number of fire-induced core damage events by the number of plant operating years. This is the approach used by a number of authors in estimating overall plant CDF following the March 11, 2011 Fukushima Dai-ichi reactor accidents [43-45]. In the case of fire, since there have been no fire-induced core damage events, a conventional PRA approach is to, as described in NUREG/CR-6823 [46], either: a) compute a chi-squared upper bound, or b) to perform a Bayesian update on a non-informative, Jeffreys prior distribution where the evidence consists of zero events over the industry history. A third approach, discussed further in the following section, uses estimates of the Conditional Core Damage Probability (CCDP) derived for operational fire events as data.[m]

All of these statistical approaches are based on two important simplifying assumptions: a) the plants in the analysis group are nominally identical, and b) the plants have not changed over time.[n] The first assumption discounts one of the key lessons from PRAs over the years: risk is plant specific [47]. Regarding the second assumption, U.S. plants have made numerous fire-safety related improvements in response to events and associated regulatory actions (e.g., the promulgation of Appendix R to 10 CFR 50.48 [48] following the Browns Ferry fire) and analyses (e.g., the IPEEEs).

To illustrate the potential importance of these two assumptions (referred to by Apostolakis under the unifying title of "exchangeability" [49]), Figure 2 shows a distribution of recent point estimates of total CDF (i.e., the CDF from all contributors) obtained from 41 risk-informed LARs. These LARs cover 61 units and multiple application areas, including allowed outage times, surveillance frequencies, integrated leak rate testing, and reactor power uprates, as well as NFPA 805. The LARs were submitted over the period 2002 through 2013; over 75% were submitted after 2007. Figure 3 shows how recent estimates for total CDF compare against estimates derived from the IPE and IPEEE studies.[o] (Each point corresponds to a single unit.) Figure 2 shows that there are order-of-magnitude variations in the CDF estimates.[p] Figure 3 shows that most (but quite not all) of the total CDFs have decreased, some by a substantial amount. As discussed in numerous plant-specific supplements to NUREG-1437 [50], these changes can be attributed to both PRA modeling changes and the incorporation of actual improvements in plant design and operation.[q]

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[m] *The CCDPs are conditional probabilities of core damage, given the degraded plant conditions (e.g., equipment failures and human errors) experienced during the event. Broadly speaking, they can be viewed as measures of "how close" each event came to core damage.*

[n] *Although more sophisticated statistical analysis techniques are available to address heterogeneity within a group and time dependence, such methods require more data than available.*

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

[p] *It is important to recognize that the LAR estimates were developed for varying purposes and are of varying vintage. As discussed in a recent Nuclear Energy Agency survey report [7], variations in PRA results can be due to a number of reasons, including differences in: the purpose of the PRA, the PRA scope and level of detail, and the "maturity" (we would argue "realism") of analyses of different accident classes and contributors. Figure 2 provides a rough indication of U.S. CDFs but not a definitive picture of sufficient quality to support specific regulatory decisions.*

[q] *Supplements 2 through 55 of NUREG-1437 [50] provide, among other things, NRC staff evaluations of licensee estimates of CDF and LERF provided to support assessments of potentially cost-effective Severe Accident Mitigation Alternatives (SAMAs) for plants applying for license extensions. A number of the estimates come from full-scope licensee PRAs; others involve a mixture of estimates from a variety of sources (e.g., the plant's current internal events PRA and its IPEEE analysis), perhaps modified to address concerns raised during the staff evaluations. Many of the staff evaluations include considerable historical information indicating how the plant's PRA changed over time to reflect both analytical and actual changes.*

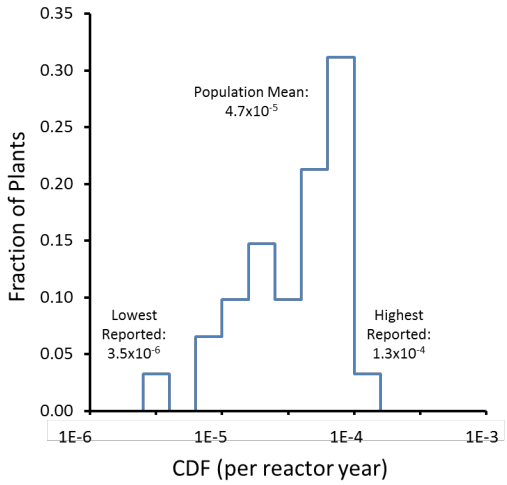

Figure 2. Recent total CDF point estimates

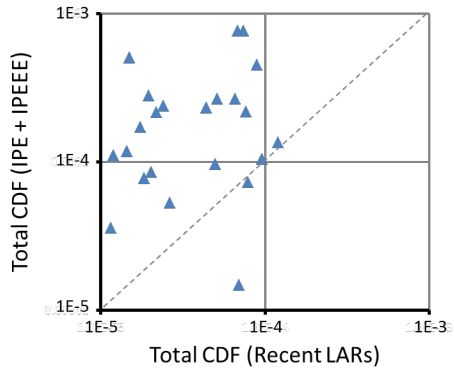

Figure 3. Comparison of recent and past CDFs

Apostolakis [49] states:

> *"The need for the assumption of exchangeability between past, present, and future reactors makes global statistical estimates of little value in regulatory decision making."*

From a regulatory decision support perspective, this caution is critical. Indeed, in 1983, for the purposes of the Indian Point hearings, the ASLB cited the importance of plant changes when rejecting the use of plant-level, historical information in estimating the CDFs for the Indian Point plants [36]. PRAs remain the best source of information for an integrated view of the effects of plant design and operation on risk. However, for the purpose of exploring the extent to which available data support (or deny) arguments regarding the conservatism of fire PRA, we think it informative to perform a limited statistical analysis, as discussed below.

4.1.2 Empirical Estimates of Fire CDF Based on Plant-Level Operating Experience

Although numerous fires have occurred in U.S. nuclear power plants, the vast majority of these events have not been of high safety significance. Out of the 1695 fire events for the period 1990-2009 included in the EPRI Fire Events Data Base (FEDB), only 28 were classified as "challenging," and this designation is based on a judgment that the fire had a substantive effect on the environment outside the initiating source, not the nature or significance of the components actually affected [51,52]. A review of operational events reported to the NRC via Licensee Event Reports (LERs) per the requirements of 10 CFR 50.73 [53] indicates that in the period 1985-2012, 110 of these events have been initiated by (or later involved) fires [54].[r] Most of these events did not represent major challenges to nuclear safety: none were classified

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

by the NRC's Accident Sequence Precursor (ASP) program as "significant" (with CCDPs greater than 1E-3) and only two had CCDPs between 1E-4 and 1E-3.[s]

To explore what precursor events can tell us about fire CDF, we follow the approach of Gallucci, who uses event CCDPs as data points [56]. In particular, Gallucci's approach involves using the sum of CCDPs, a non-integer value, in place of the number of events in conventional statistical estimation processes. This approach is similar to early analyses using operational experience when estimating CDF [36,57] and to current PRA treatments of "impact vectors" in common cause failure analysis [58].

Considering precursor events occurring over the period 1969-2004 (see Table 4), Gallucci estimates that the fire CDF for a U.S. plant is 7.1E-5/ry. We note that:

- with the exception of the Browns Ferry fire, all of the events in Table 4 occurred after the promulgation of Appendix R in late 1980; and
- a notable 1968 electrical cable fire at San Onofre 1, which affected a number of important systems, is not included in Table 5. This event pre-dated the earliest PRA studies and no CCDP was estimated. Further, as discussed in NUREG/CR-6738, the plant was constructed prior to the development of current cable flammability standards [42].

In the period 2005-2014, two fire-related precursor events with CCDPs > 1E-4 have occurred in the U.S. These involved a March 28, 2010 fire at the H.B. Robinson 2 plant, and a June 7, 2011 fire at the Fort Calhoun plant (see Table 5). Neither was a "significant" (CCDP ≥ 1E 3) precursor (as designated by the ASP program).

[s] *The statuses of the ASP program and the Standardized Plant Analysis Risk – SPAR – models used to estimate event CCDPs are described in annual NRC staff papers, e.g., SECY-13-0107 [55]. The ASP program considers all events and degraded plant conditions reported in LERs. Currently an event is classified as a precursor if the CCDP for that event is greater than a set criterion (the greater of 1E-6 and the plant-specific CCDP for a non-recoverable loss of balance-of-plant systems).*

Table 4. Precursor events included in Gallucci's analysis

Plant	Date	CCDP [56][a]	Event Notes[b]
Browns Ferry 1 & 2	3/22/75	0.20[c]	Multi-unit cable fire; multiple systems lost, spurious component and system operations; makeup from control rod drive pump (non-proceduralized action)
Rancho Seco	3/19/84	2.2E-6	Main generator explosion and fire; damage to non-nuclear instrumentation power supply complicated shutdown
Oconee 1	1/3/89	3.3E-6	Reactor coolant pump (RCP) switchgear fire during power escalation; operators exceeded allowed cooldown rate
Waterford 3	6/10/95	9.1E-5	Non-safety 4 kV switchgear fire; partial loss of offsite power (LOOP)
Surry 1 & 2	10/9/99	1.2E-6 (each)	4 kV bus bar connection fire (small); loss of two emergency buses due to electrical fault
Diablo Canyon	5/15/00	9.6E-5	12 kV bus fire damaged nearby 4 kV bus; loss of offsite power to all 4 kV loads
San Onofre 3	2/3/01	1.4E-4	Switchgear fire following outage; loss of non-safety power
Quad Cities 2	8/2/01	6.6E-5	Main transformer fire following lightning strike; loss of normal offsite power
Watts Bar 1	9/27/02	3.3E-4	Offsite (hydroelectric station) fire; LOOP, fire brigade dispatched offsite, reduced onsite staffing

[a] A number of the CCDP estimates differ from those provided in SECY papers providing the status of the ASP program (e.g., [59,60]). Also, the Watts Bar Hydroelectric Station fire is not relevant for our discussion. However, as pointed out by Gallucci, the analysis results are dominated by the CCDP value assigned to the Browns Ferry fire; the numerical values of the other CCDPS have a minor effect.
[b] See NUREG/KM-0002 [61] for a recent compilation of information on the Browns Ferry fire and NUREG/CR-6738 [42] for a fire-PRA oriented discussion of that event. The notes on the remaining events are based on their LER Summaries.
[c] This value is consistent with SECY-10-0125 [62]. Gallucci notes other estimates range from 0.03 to 0.40.

Table 5. Fire precursors with CCDP > 1E-4, 2005-2014

Plant	Date	CCDP [55]	Event Notes
H.B. Robinson 2	3/28/10	4E-4	4 kV cable fire, loss of RCP seal cooling and additional equipment failures; operators fail to diagnose plant conditions and control plant; operator actions cause a second fire
Fort Calhoun	6/7/11	4E-4	480V switchgear fire during cold shutdown; loss of multiple safety buses (combustion products migrated to non-segregated bus duct)

To update Gallucci's analysis to incorporate the new evidence from the Robinson fire (but not the Fort Calhoun fire since that event occurred during cold shutdown) and address uncertainties (Gallucci focuses on point estimates), we perform three case studies employing a Bayesian estimation process.

Case 1: The prior distribution is based on all events in Table 4 and the number of U.S. reactor years through 2004. The updating evidence consists of the Robinson CCDP shown in Table 5 and the U.S. reactor years from 2005 through 2014.[t]

Case 2: The prior distribution is based on the experience from 1980 (i.e., the promulgation of Appendix R) through 2004. The updating evidence is the same as in Case 1.

Case 3: This is the same as Case 1, except the Browns Ferry CCDP is set to 0.02 (i.e., slightly lower than the lowest CCDP estimate of 0.03 identified by Gallucci[u] and one-tenth of the value used in Case 1).

In all cases, the prior distribution is constructed as a "constrained non-informative prior distribution" [46] to represent a relatively diffuse state of knowledge.

The numerical results of our analysis are shown in Table 6 and Figure 4. (Table 6 also shows the results of a conventional PRA treatment of the available statistical evidence, i.e., zero fire-induced core damage events through the end of 2014 using the previously mentioned Jeffreys prior and chi-squared approaches.) The details of our analysis (including the analytical methodology and the numerical inputs) are provided in Appendix B. Key limitations of our analysis are discussed in Section 4.1.3.

It can be seen that the practical effect of reducing the Browns Ferry CCDP by a factor of 10 (from Case 1 to Case 3) is to shift the posterior distribution by a factor of 10. This is due to the dominance of the Browns Ferry fire evidence (it overwhelms the combined effect of the remaining events), even after the factor of 10 reduction.

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[t] *Our calculations neglect the effect of reactor downtime. This effect is small relative to the uncertainties associated with more fundamental assumptions discussed later in this paper.*

[u] *This estimate was developed by Apostolakis and Kazarians a few years after the Browns Ferry fire [63].*

Table 6. Comparison of CCDP-based estimates for fire CDF (/ry)

	Mean	5th	50th	95th
Precursor Case 1 (Update of Gallucci analysis)	6E-5	3E-7	3E-5	2E-4
Precursor Case 2 (Discounting Browns Ferry)	3E-7	1E-9	1E-7	1E-6
Precursor Case 3 (Browns Ferry CCDP = 0.02)	7E-6	3E-8	3E-6	3E-5
Zero Events, Jeffreys Prior[a]	1E-4	5E-7	6E-5	5E-4
Zero Events, Chi-Squared "Upper Bound"[a]	5E-4			

[a] Evidence is zero fire-induced core damage events in 3939 reactor years

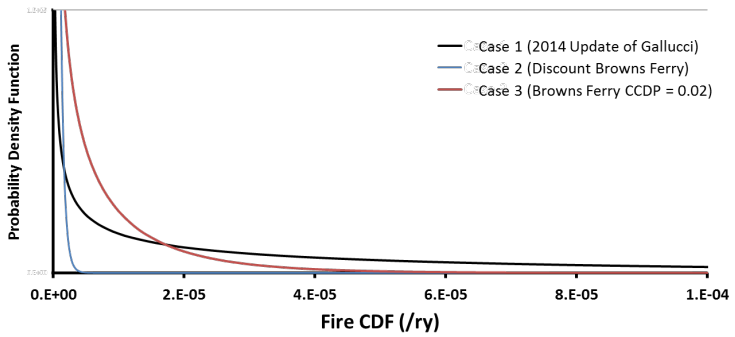


Figure 4. Comparison of fire CDF distributions (/ry)

Recognizing that U.S. plants have made numerous fire-safety related improvements in response to events and associated regulatory actions (e.g., the promulgation of Appendix R in late 1980) and analyses (e.g., the IPEEEs) since Browns Ferry, it might be argued that Case 1 is not representative of the current situation. While not disagreeing with the sentiment, we caution that, given the sparseness of accident data, discarding data is something to be done with extreme care. For example, although one of the prime lessons from the Browns Ferry fire was that water should be used to promptly extinguish electrical fires, the reluctance to use water contributed to delayed fire suppression in a 1995 fire event [42]. Recently, such reluctance was echoed in remarks made during a Commission hearing on fire protection [64]. Our three cases cover the range of views on the applicability of the Browns Ferry event and the CCDP of that event.

Table 6 and Figure 4 show that the CCDP-based estimates are extremely uncertain. (The "reverse-J" shaped distributions in Figure 4 indicate that very small values cannot be ruled out.) This is not surprising since the evidence consists only of CCDPs and not actual events, and all of the non-Browns Ferry CCDPs are very small (on the order of 1E-4 or less); their sum (including the Robinson event) is approximately 1E-3.

Table 6 and Figure 4 also show that the inclusion of pre-Appendix R experience (including the Browns Ferry fire) appears to make a qualitative difference in our comparison: the distribution for Case 2 is significantly left-shifted from those for Cases 1 and 3. Because the Browns Ferry fire is the only event with a significant CCDP, this observation is also not surprising.

4.1.3 Comparing Empirical and Fire PRA Estimates

To return to our question regarding what available plant-level evidence can tell us about the realism of current fire PRAs, we wish to compare the precursor-based estimates derived in the previous section with estimates derived from the PRAs.

Table 7 and Figure 5 compare the results of the three cases with fire CDF estimates from NFPA 805 LAR submittals.

It is important to recognize that the comparison is done on an industry-wide basis – the figure shows the total U.S. fire CDF (i.e., the sum of all the individual plant fire CDFs) estimated using the precursor CCDPs and using the NFPA 805 LAR estimates. This metric, denoted by $F\text{-}CDF_{US}$, quantifies the potential of observing a fire-induced core damage event somewhere in the U.S. fleet. This is to be contrasted with the fire CDF produced by a plant-specific fire PRA; the latter quantifies the potential of observing a fire-induced core damage event for the plant being studied. We use the industry-wide approach to support an "apples to apples" comparison (the precursor-based estimate addresses an "average plant," whereas the LAR estimates are plant-specific) and to facilitate comparisons with total U.S. operating experience, similar to those performed by Canavan et al. [18].

The details of our analytical methodology are provided in Appendix B.[v]

Table 7. Comparison of Estimates for $F\text{-}CDF_{US}$ (/yr)

	Mean	5th	50th	95th
Precursor Case 1 (Update of Gallucci analysis)	6E-3	2E-5	3E-3	2E-2
Precursor Case 2 (Discounting Browns Ferry)	3E-5	1E-7	1E-5	1E-4
Precursor Case 3 (Browns Ferry CCDP = 0.02)	7E-4	3E-6	3E-4	3E-3
Fire PRA (NFPA 805)	5E-3	2E-3	5E-3	7E-3
Zero Events, Jeffreys Prior[a]	1E-2	5E-5	6E-3	5E-2
Zero Events, Chi-Squared "Upper Bound"[a]	5E-2			

[a] Evidence is zero fire-induced core damage events in 3939 reactor years

[v] *Based on the results of some uncertainty analyses in early full-scope PRAs, the analysis assumes that the uncertainty in each fire CDF is characterized by a lognormal distribution with an error factor of 10. Table 7 and Figure 5 show that the error factor for $F\text{-}CDF_{US}$ is largely due to the well-known reduction in uncertainty (relative to the mean) when summing a large number of independent random variables.*

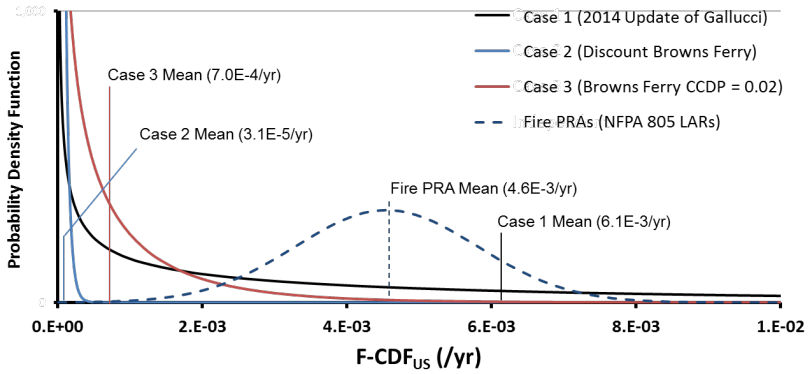


Figure 5. Comparison of precursor- and PRA-based distributions for F-CDF$_{US}$

Similar to our discussion in the preceding section, Table 7 and Figure 5 show that the treatment of Browns Ferry makes a qualitative difference in our comparison of fire PRA results with empirical evidence. If Browns Ferry is included (Case 1), the mean of the fire-PRA based estimate for F-CDF$_{US}$ is similar to that developed from the CCDPs. If Browns Ferry is not included (Case 2), most of the distribution for the fire-PRA based estimate is well above the precursor-based mean value.

In comments on an earlier version of this paper [23], we have been reminded that a number of the fire PRA estimates reported in the NFPA 805 LARs take credit for planned changes aimed at significantly reducing fire CDF. We do not have the pre-change CDF estimates, but recognize that these should be higher than the LAR estimates. This would move the fire-PRA based estimate of F-CDF$_{US}$ to the right in Figure 5.

Of course, as recognized previously, the uncertainties in the precursor-based estimates are extremely large. To investigate their effect on our comparison, we compute a summary metric that incorporates the uncertainties in F-CDF$_{US}$: the probability of observing N fire-induced core damage accidents (anywhere in the U.S.) over a fixed exposure period.[w]

Figure 6 shows the results obtained for an exposure period of 10 years. It can be seen that the differences between the Case 1 and fire-PRA based estimates are negligible. (The difference is most noticeable for N = 2, an unrealistic situation since should a core damage event actually occur, major changes to plants, the regulatory system, etc., and thereby F-CDF$_{US}$, will almost certainly result.) On the other hand, the differences between Case 2 and the fire PRA estimate are large. Similar conclusions result when the exposure period is increased to 50 years.

[w] *This probability is the Poisson distribution averaged over all possible values of F-CDF$_{US}$. Details of the approach are provided in Appendix B.*

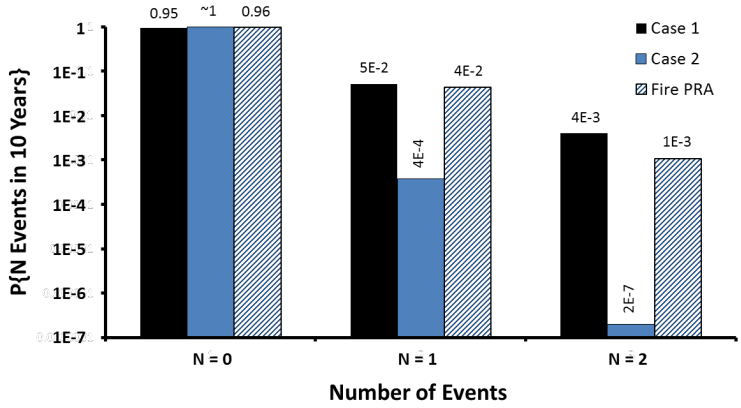


Figure 6. Comparison of precursor- and PRA-based estimates of U.S. fire-induced core damage event probabilities

We have already noted concerns with the assumption of exchangeability required by a statistical analysis, and with the lack of quantitative information for the pre-change fire CDFs of plants transitioning to NFPA 805. Additional cautions with our analysis include the following.

- Our statistical analysis:
  - is limited to precursors that involved initiating events (e.g., a plant trip or loss of offsite power – LOOP) – it does not address the CDF implications of precursors involving degraded conditions;
  - uses event CCDPs as objective data, whereas: a) the approach for estimating CCDPs has improved over time, and b) such use represents an engineering approximation to a more rigorous treatment of data uncertainty [58]; and
  - is based on precursor results that utilize a "failure memory" assumption where observed successes are modeled at their nominal failure probability and failure events are modeled as they occurred during the event. This may limit the applicability of these results for a more general PRA analysis (e.g., the full spectrum of potential fire-related damage is not considered).
- Our fire-PRA based estimate of F-CDF_{US} is based on the assumption that the fire CDFs reported in the NFPA 805 LARs are representative of those that would be generated for plants who have not yet updated their fire PRAs.

These limitations, as well as the extremely large computed uncertainties in our results, limit on our ability to draw strong conclusions from our comparison of statistical and fire-PRA based estimates. (For example, we cannot use our analysis to confidently quantify the potential degree of conservatism associated with current fire PRA technology and practices.) They also strengthen the need to assess the realism of fire PRA from a variety of perspectives, as we do in the following sections.

However, these limitations do not mean that we should ignore the qualitative lessons from our comparison. Consider the proximity of the mean estimates for F-CDF_{US} from Case 1 and the NFPA 805

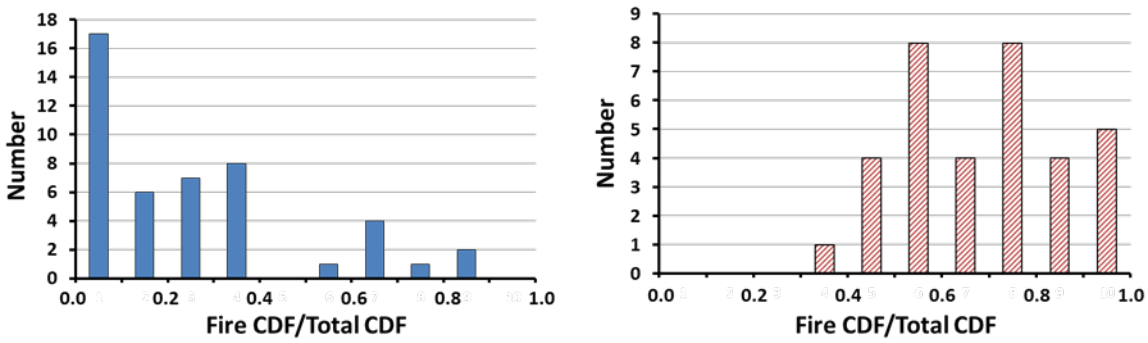
LARs. (Recognizing the large uncertainties inherent in PRA results, these two estimates are nearly the same value.) Turning the exchangeability discussion around, this proximity suggests that in order for the fire PRA estimate to be consistent with plant-level operational experience, either Browns Ferry would have to be exchangeable with later events, or the CCDPs for at least one later fire event would have to have been underestimated by several orders of magnitude. Neither condition is appealing. Recognizing that the post-change fire CDFs used in our analysis are likely smaller than the pre-change fire CDFs, it would appear that, as contended by a number of authors, current fire PRAs are providing conservative results (at least when considered at a national aggregate level).

As a second qualitative lesson, we observe that the strong sensitivity of our results to the treatment of pre-Appendix R experience, including Browns Ferry, reinforces the importance of ensuring that the lessons learned from that event (and, by extension, other major fire events) are retained and acted upon.

4.1.4 Relative Contributions to Total Plant CDF

The preceding analysis uses available operational experience but requires a number of major assumptions, the most important one being that of event exchangeability. To provide a second, but still CDF-based perspective on the realism of current fire PRAs, we look at past and current estimates for the relative contribution of fires to the overall CDF.

Figure 7 compares the relative contribution of fire to total CDF from IPE/IPEEE studies (mainly performed in the mid-late 1990's) and from NFPA 805 applications. The IPE/IPEEE estimates come from the 46 units (at 28 plants) which either completely screened seismic events (implying a small seismic CDF) or developed seismic CDF estimates as part of their IPEEE analyses. Most of the NFPA 805 estimates (for 34 units at 21 plants) come from the original LAR or responses to NRC Requests for Additional Information (RAIs). A few estimates come from staff evaluations of the licensee submittals. Figure 8 compares the ratio of fire CDF to internal events CDF for the IPE/IPEEE studies (98 units) and for the same set of NFPA 805 submittals addressed in Figure 7b.



(a) IPE/IPEEE **(b) NFPA 805**

Figure 7. Fire contribution to CDF: comparison of IPE/IPEEE and NFPA 805 LAR estimates

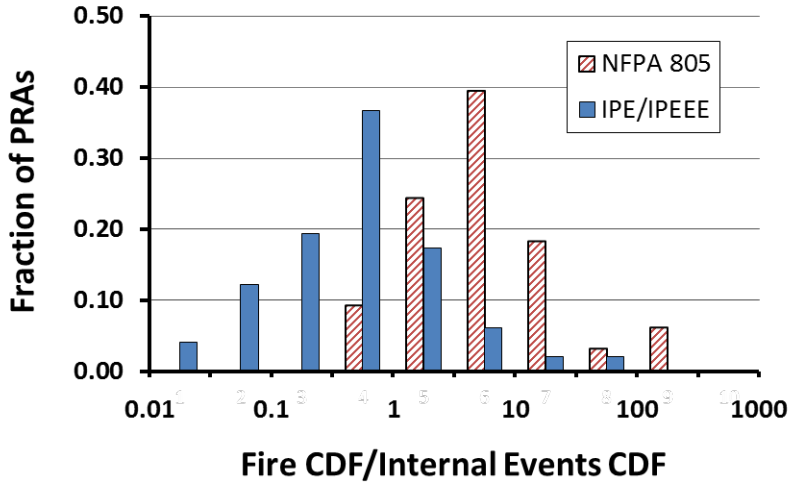


Figure 8. Ratio of fire CDF to internal events CDF: comparison of IPE/IPEEE and NFPA 805 estimates

Recognizing that the recent LAR submittals represent a smaller sample, nevertheless the difference between the two sets of results is striking. In the IPE/IPEEE studies, fire is an important contributor for many plants. In the recent LAR submittals, fire is a major or even dominant contributor for most plants. Possible explanations for this change include: a) the numerous plant changes made since the IPE/IPEEE studies[x] were preferentially effective for non-fire related initiators, b) the IPEEE studies underestimated the importance of key issues addressed in the recent studies (we discuss changes in current understanding of fire and fire PRA technology in Section 4.3 of this paper), or c) the recent fire PRA results are indeed conservative.

It is interesting to observe that in the last 5 years, there have been two precursor events with a CCDP greater than 1E-4: the 2010 Robinson fire (see Table 5) and the 2011 North Anna earthquake [65]. This observation supports the viewpoint that fire continues to be an important contributor to CDF, and does not contradict the viewpoint that fire could be a dominant contributor.

4.1.5 Industry Analyses

In 2010, Canavan et al. performed an analysis that compared fire PRAs with operational experience [18]. (The results of this analysis have been more recently discussed by Chapman [66].) The analysis did not attempt to perform a CDF-level statistical analysis, but looked at some of the constituent parts (including intermediate outputs) of a fire PRA, illustrating key points using a number of fire PRAs performed to support fire protection program changes per NFPA 805. The analysis raised concerns regarding model input (fire frequencies and severities, as represented by heat release rates), and calculated consequences (the frequency of spurious operations and the frequency of severe safety challenges as represented by high CCDPs).

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[x] *Descriptions of post-IPE/IPEEE plant changes and associated changes in estimated CDF are provided in many of the plant-specific supplements to NUREG-1437 [50].*

Canavan et al. raise a number of good points regarding the cumulative effect of multiple modeling conservatisms (each of which may not have a major impact individually). However, as a counterpoint to these points, we note that:

- the analysis' concern with the overestimation of the frequency of fires, recently echoed by Saunders and Burns [67] is based on a recently disproven hypothesis of a systematic downward trend in fire occurrences (which leads to a discounting of older events in the fire database) [68];
- some of the concerns address fire PRA outcomes that are not the focus of the analysis, and are typically not, by themselves, measures that analysts would normally consider when deciding if further iteration is needed; and
- it is unclear whether addressing the conservatisms identified would be cost-beneficial. (For example, EDGs were apparently unimportant for six of the seven plants considered in the analysis. It is conceivable that the resources required to perform more realistic analysis would not result in significantly different results.)

These observations aside, we find the authors' argument regarding the over-prediction of events with high CCDPs to be compelling, and only observe that their results could be due to conservatisms in the estimation of the CCDPs (e.g., due to scoping-level assessments of the plant impact of fire damage or to the neglect of recovery actions) as well as conservatisms in the estimation of the frequencies of scenario-specific fire damage zones.

4.2 Important Scenarios

In discussions of the benefits of PRA, we often hear statements along the lines of "the PRA numbers (i.e., output measures such as CDF and LERF) are overemphasized – the true value of PRA is its ability to provide risk insights." Although this position is arguable from a practical point of view (after all, the quantitative results of PRA play a prominent role in current risk-informed regulatory applications and often highlight key insights), there is no debate concerning the intrinsic value of PRA's more qualitative outputs. Clearly, the PRA's identification of the major contributors to risk, including their key characteristics, helps us determine what can be done to affect these contributors and is important for efficient and effective risk management. In addition, the contributors identified by a PRA also provide a means for us to understand the PRA model and to test that model against other sources of information, thereby helping us assess our confidence in the PRA results.

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

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

4.2.1 Fire PRA scenarios

Past studies (including the IPEEEs), taken as a whole, have consistently found that fires involving electrical cables and/or cabinets in key plant areas (e.g., main control rooms, emergency switchgear rooms, cable spreading rooms, cable vaults and tunnels) are the dominant contributors to fire risk [1-5,69,70]. In a number of these areas, the risk-significant scenarios can involve fires that start in

electrical cabinets but propagate to cables outside. Typically, the fire effects are relatively localized (i.e., not room-encompassing) – the fire is important because it affects a local concentration of important cables. However, the IPEEEs have shown that for some plants, large turbine building fires and fires inducing main control room abandonment could be important.[y]

The risk-significant accident sequences triggered by fires are generally dominated by some form of transient (e.g., loss of feedwater, LOOP, loss of various support systems) but loss of coolant accidents (LOCAs), including reactor coolant pump (RCP) seal LOCAs and transient-induced LOCAs involving stuck open relief valves are important for some plants. Scenarios involving non-fire related failures can be visible contributors to risk, but the risk tends to be dominated by scenarios in which the initiating fire causes enough damage to cause core damage directly (if such scenarios exist for the plant being analysed) [69,71].

The technical lessons stemming from recent fire PRA studies have not yet been synthesized as many of the NFPA 805 submittals are undergoing NRC review. To shed some light on important scenarios, we consider the results of NRC's SPAR-AHZ models[z] [55,72]. The three most recent models (all for PWRs) address fire scenarios using information from NFPA 805 submittals. These models are benchmarked against the licensee models; the differences are not important for the purposes of this paper.

The important scenarios identified by the three SPAR-AHZ fire models are, for the most part, consistent with those identified in past studies. Electrical fires in the usual important areas (e.g., main control rooms, cable rooms, switchgear rooms) and turbine building fires are important contributors at some or all of the plants. The plant response scenarios triggered by these fires typically involve some form of transient (including LOOP scenarios), sometimes involving the spurious opening of a power-operated relief valve (PORV). Fire-induced RCP seal LOCAs are not important contributors at these plants.

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

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

- The total mean frequency of scenarios involving reactor trip (automatic or manual) ranges from 0.06/ry to 0.30/ry. (As with past fire PRAs, it is assumed that every unscreened fire scenario results in a reactor trip.)
- The total frequency of scenarios involving fire-induced LOOP ranges from 8E-3/ry to 1.5E-2/ry. These LOOPs are modelled as being unrecoverable.
- Scenarios involving main control room abandonment (MCR) are not major contributors, ranging from 0.1% to 2% of total fire CDF. The CCDPs for these scenarios span a large range, going from 0.06 up to 1.0. For plants with higher CCDPs, it can be seen that the low CDF contribution

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[y] *In contrast with U.S. analyses, many international fire PRAs assume that a severe fire will cause the loss of the entire room and emphasize the analysis of multi-compartment scenarios.*

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

is due to the estimated low frequency of fires spurring evacuation, not the modeled robustness of the plant response.

- The fire PRA models generate thousands of detailed event sequences that need to be quantified. (By comparison, the older SPAR-EE models generate on the order of 50 sequences to be quantified, more for models if MCR scenarios are divided into cabinet-level sub-scenarios.) This creates challenges not only for the software quantification tools, but also more subtle challenges for model checking during model development and after quantification.

4.2.2 Observed fire scenarios

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

Table 8 provides brief summaries of the 1975 Browns Ferry fire and five non-U.S. fire events involving multiple safety system losses and serious challenges to core cooling identified and analysed in NUREG/CR-6738 [42]. It is important to recognize that none of these events involved plants of U.S. design, and that the latest event occurred in 1993; we are unaware of any severely challenging fires since the Narora fire.

The events identified in Tables 4, 5, and 8 represent a very small fraction of the fire events that have occurred. For the U.S. alone, the EPRI Fire Events Database includes reviewed records for nearly 1700 fire events occurring over the period 1990 through 2009 [51,52]. However, the vast majority of these events have posed minor challenges to nuclear safety and are not addressed in our current, high-level analysis. (An integrated review of these events similar in spirit to that done in NUREG/CR-6738 would likely be useful in an analysis of fire PRA modeling of intermediate, pre-core damage plant states.)

4.2.3 Comparison of fire PRA and observed scenarios

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

The one major potential concern arises from the high risk importance given to switchyard/transformer yard fires by the three SPAR-AHZ models (and the associated licensee NFPA 805 models). Such fires (including large station transformer fires, have been reported – a recent review of significant fire incidents [54][aa] identifies 50 relevant events in the 1985-2012 time period – but none have been assessed to be significant precursors. We observe that switchyard/transformer yard fires appear to be visible CDF contributors in two of the seven fire PRAs reviewed by Canavan et al. [18] but do not know if the potentially anomalous estimated importance of such fires is due to fire PRA technology limitations, analyst-driven simplifications, or aspects of the NFPA 805 licensing process.

[aa] *Ref. 54 classifies a fire event as "significant" if the event was reported via the requirements of 10 CFR 50.72(a)(1)(i) or would likely have required reporting had plant personnel not taken action. The 50 events referenced: a) occurred during power operation, and b) involved Switchyard, Transformer/Transformer Yard, and other outdoor (but onsite) fires.*

Table 8. Fires Involving Severe Challenges to Core Cooling

Plant	Type	Date	Summary Description (based on narratives provided in [42])
Browns Ferry 1 & 2	BWR	3/22/75	Multi-unit cable fire; multiple systems lost, spurious component and system operations; makeup from control rod drive pump (non-proceduralized action)
Greifswald 1	VVER-440	12/7/75	92 min cable fire in or near 6 kV switchgear started by electrical fault; caused station blackout (SBO), loss of all normal core cooling for 5 hours, loss of coolant through pressurizer safety (failed to reclose); recovered through low pressure pumps and cross-tie with Unit 2 to power one AFW pump (non-proceduralized actions).
Beloyarsk 2	LWGR -1000	12/31/78	Lube oil fire in turbine building collapsed turbine building roof, propagated into several elevations of control building (open penetrations, cable shafts); damaged main control room (MCR) panels; secondary fire from oil-filled transformer; extreme cold weather, fire under control in 17 hours, extinguished in 22 hours; damage to multiple safety systems and instrumentation, reactor control was "extremely difficult."
Armenia 1 & 2	VVER-440	10/15/82	Short circuit in a 6 kV cable led to 7 fires (ignition points) in 2 different cable galleries; fire spread to other cables, smoke spread to several areas including Unit 1 MCR; Unit 2 had some lesser fire effects; automatic foam system in manual and not actuated, fire brigade did not start attack for 20 minutes because power was on; fire under control in 6 hours, extinguished 1 hour later; Unit 1 SBO caused by hose streams about 2 hours into fire, loss of instrumentation and reactor control about 1 hour later; event also involved secondary H_2 explosion, lube oil fire and transformer explosion; recovered via temporary cable from emergency diesel generator (EDG) to high pressure pump (non-proceduralized action), recovery of Unit 1 MCR power from Unit 2 sources.
Chernobyl 2	RBMK -1000	10/11/91	Turbine failure, H_2 and oil release, large turbine building fire, turbine building roof collapsed causing loss of generators, eventual loss of all feedwater (direct damage from falling debris or de-energization to aid local fire fighting; feedwater supply intermittent, operators use seal water supply system for makeup; reactor control regained in 3.5 hours, recovery actions well outside written and practiced procedures; fire put under control also in 3.5 hours, fire extinguished 2.5 hours later.
Narora 1	PHWR	3/31/93	Turbine blade failure, H_2 explosion and fire, large turbine building fire; fire propagated along cable trays, smoke forced abandonment of MCR (shared between units) for 13 hours, fire caused loss of power to Unit 1 shutdown panel but not Unit 2 shutdown panel; major part of fire put out in 1.5 hours, fire fully extinguished 7.5 hours later; SBO 10 minutes into event, diesel-driven fire pumps used to feed steam generators, tripped by apparent common-cause (but not fire-related) failure 3.5 hours later, one pump restarted 1.75 hours later; operators "flying blind" for 4.5 hours (until staff entered containment to read instruments); EDG started and loaded 5.5 hours into event but shutdown cooling pump not energized until 17 hours – this led to declared end of SBO conditions.

A somewhat lesser potential concern is revealed by a more quantitative look at the intermediate results of the three SPAR-AHZ models. As indicated earlier, these results suggest a high rate of fire-induced reactor trips (on the order of 0.1/ry) and fire-induced LOOPs (on the order of 0.01/ry). Reviewing the LERs for 1980-2012, it appears that the U.S. average rates (based on around 80 fire-related trips and seven fire-related LOOP events in that time period) are on the order of 0.03/ry and 2E-3/ry, respectively. At this point, we do not know if this apparent conservatism applies to a broader set of current fire PRAs. Also, as discussed earlier in this paper, conservatism in estimated intermediate state frequencies does not

necessarily imply conservatism in CDF estimates. However, even less-than-order of magnitude mismatches between the model estimates and empirical experience can erode confidence in the models.[bb]

Regarding our second point of comparison, it appears that most of the events listed in Tables 4, 5, and 8 represent, at a high level, scenarios involving fire sources and induced transients typically included in fire PRAs. (The 1989 Oconee fire, which led to an overcooling transient with a potential challenge to reactor pressure vessel integrity, may be an exception.) However, it also appears that the U.S. precursors have involved a number of features not addressed in current fire PRAs:

- multiple fires (e.g., the 2010 Robinson event);
- multiple hazards (e.g., the 1984 Rancho Seco fire where debris from the hydrogen explosion appears to have been the principal cause of damage);
- fires as consequences rather than initiators of a scenario (e.g., the second fire during the 2010 Robinson event); and
- offsite influences on onsite fire protection activities (e.g., the 2002 Watts Bar fire, in which the plant's response to an offsite fire degraded the plant's capability to mitigate onsite fires).

These observations echo a number of points made by NUREG/CR-6738 [42] in its detailed review of 30 notable fire events (including the events summarized in Table 8). NUREG/CR-6738, which was specifically intended to identify potential areas for fire PRA technology improvement based on lessons from operational events, states that "the overall structure of a typical fire PRA can appropriately capture the dominant factors involved in a fire incident" but also notes several modeling challenges. These include the treatment of:

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

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

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

[bb] *Whether such a difference* ***should*** *affect user confidence, given the uncertainties in PRA modeling and results, is a point for discussion within the broader PRA community.*

4.3 Fire PRA Analysis Technology

The preceding section focuses on the results of current fire PRAs. This section briefly discusses the history of NRC-sponsored fire PRA research and development (R&D) and the current status of efforts aimed at improving the methods, models, tools, and data available for fire PRA. Additional details on the history of NRC's fire-protection related R&D (including fire PRA R&D) can be found in NUREG/BR-0364 [73]; copies of related NUREG and NUREG/CR reports covering from 1976 through the early portion of 2013 can be found on the DVD provided with NUREG/KM-0003 [74].

4.3.1 Historical Development

The March 22, 1975 fire at Browns Ferry prompted the NRC to perform a limited study of the risk significance of that event, published as a supplement to WASH-1400 [9]. The study indicated that the CDF associated with that fire was around 1E-5/yr, or about 20% of the CDF due to causes addressed in the main body of WASH-1400 (e.g., LOCAs, plant transients). The study also noted the usefulness of developing a more detailed fire PRA methodology (including improved models and data).

In 1977, the NRC initiated a research project on fire risk. The objective of the project, led by Apostolakis at the University of California at Los Angeles (UCLA), was to develop a methodology for estimating fire risk at nuclear power plants. The project was intended to complement NRC's Fire Protection Research Program, initiated in 1974, which was being performed at Sandia National Laboratories [73,75]. The UCLA project was later supplemented by additional NRC-sponsored efforts at the Rensselaer Polytechnic Institute (e.g., [76,77]).

Apostolakis and his team developed a methodology that was used in the industry-sponsored Zion (1981) and Indian Point (1982) PRA studies [8,1,47]. The methodology, which was documented in numerous papers and reports (e.g., [40,41,71,78]), provided the basic framework for most subsequent fire PRAs in the U.S., whether performed by industry or the NRC. As compared with an earlier, event-tree based approach [79], the methodology explicitly incorporated the predictions of deterministic models for fire behavior ("fire models") using a competing risks approach, in which the probability of fire damage to equipment (including electrical cables) was computed as the probabilistic outcome of a "race" between two simultaneous processes: fire growth and fire suppression. This approach enabled the treatment of scenario-specific factors governing the rate and extent of fire development.[cc]

In the late 1980s and early 1990s, the NRC's NUREG-1150 [2] and the industry's IPEEE studies [4,5] (performed using guidance developed by EPRI [82,83]) confirmed that fire could be an important contributor to risk. However, given the difficulty of accurately modeling fire behavior and the sparsity of empirical data for key factors (e.g., the likelihood of occurrence of potentially severe fires), it was generally considered that the technology for addressing fire risk was less developed than that for addressing a number of other initiators. (See, for example, [84].)

In 1998, RES reviewed the results of previous NRC fire-related R&D activities [85] and initiated a fire PRA R&D program to address then-current limitations. The efforts of this program were guided by a structured identification of potential R&D topics and sub-topics (see Appendix D) [86], and an evaluation of these topics considering agency needs (including deterministic as well as risk-informed) and available

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[cc] *The methodology also included practical approaches to address a number of topics that remain of interest in current PRAs, including the treatment of population variability using hierarchical Bayesian methods [79], and the quantification of model uncertainty using so-called "error factors" [80].*

resources. The identification process was supported by experiences from the IPEEE program, NRC-sponsored reviews of fire R&D issues [87,88], and feedback from the ACRS [84].

Following the completion of its initial fire PRA R&D tasks, RES intended to perform a "fire risk requantification study" for selected plants following the completion of its initial tasks [89,90]. In addition to providing improved CDF estimates for these plants, it was expected that such a study would provide implicit guidance for performing fire PRA, help identify those areas where the fire PRA improvements would most affect the fire risk results, provide experience concerning the practical implementation of these improvements, and provide insights concerning the reliability of previous fire PRA methods and tools. The events of September 11, 2001 and NRC's subsequent response drew considerable resources away from its ongoing programs, including the fire risk research program, and a number of planned fire R&D activities were not carried forth to completion. Nevertheless, the concept of a requantification study was considered to be worth pursuing, and formed the basis for discussions with EPRI on collaborative fire R&D. Eventually, these discussions led to activities resulting in the joint RES/EPRI development of EPRI 1011989/NUREG/CR-6850 [13] in 2005.

4.3.2 Current Status

Recent evaluations of the status of fire PRA technology based on NFPA 805 applications have been provided in 2011 by Stetkar et al. [19] (see Table 9, which provides a list of Frequently Asked Questions[dd] – FAQs – as well as comments on the current status of their resolution) and Gallucci [21], and in 2013 by NEI [20]. The fire PRA technical issues identified by NEI are shown in Table 10.

Tables 9 and 10 provide some information on the current state of resolution of the various issues.[ee] It is beyond the scope of this paper to provide a detailed discussion of these issues. However, we observe that all of these topics, as well as most of the topics identified by earlier authors, have been or are being addressed by ongoing work. Thus, it seems clear that progress towards improved realism is being made. Furthermore, it is important to recognize that the topics represent specific aspects of fire PRA; the overall framework and approach is not being challenged.

Of course, as with any technical field regardless of its state of maturity, there are areas for improvement. Potential R&D topics include a number of the important (but admittedly extremely difficult) operational experience issues identified in NUREG/CR-6738, particularly multiple fires, multiple hazards, and non-proceduralized actions. We also note that none of the current work appears to be explicitly aimed at developing screening tools to address what appear to be anomalous qualitative results (e.g., the risk importance of switchyard/transformer yard fires).

[dd] *The NFPA 805 FAQ process was established by NRC to provide staff guidance on a variety of NFPA 805 transition issues [91]. The FAQs listed in Table 9 are associated with the fire PRA methods and models discussed in NUREG/CR-6850 [13]. Interim responses on the FAQs are provided in NUREG/CR-6850, Supplement 1 [14]. Per Stetkar et al., as of the writing of their report, the NRC staff considered all of the FAQs to be closed, but industry considered most to be either open or only partially resolved [19].*

[ee] *As a part of the transition to 10CFR50.48(c), licensees transitioning to NFPA 805 raised a number of fire PRA questions while performing their fire PRAs. These questions became part of the overall NFPA 805 frequently asked questions (FAQ) program established by the NRC. NRC, EPRI, and the broader industry participate in this FAQ program to develop interim responses to these questions. In these tables, the indicated status of each issue reflects the resolution state and location where additional information can be found.*

Table 9. List of fire PRA FAQs, 2011 (1 of 2)*

FAQ	Issue / Concern	Current Status
Ignition Source Counting Guidance for Electrical Cabinets (FAQ 06-0016)	Need to clarify the guidance on counting electrical cabinets and panels for NFPA 805 transition applications	**Resolved** - Chapter 3; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Ignition Source Counting Guidance for High-Energy Arcing Faults (HEAF) (FAQ 06-0017)	Need to clarify the guidance for counting HEAFs associated with switchgear and load centers	**Resolved** - Chapter 4; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Ignition Source Counting Guidance for Main Control Board (MCB) (FAQ 06-0018)	Need to clarify the guidance on counting main control boards	**Resolved** - Chapter 5; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Miscellaneous Fire Ignition Frequency Binning Issues (FAQ 07-0031)	Need to clarify guidance for counting miscellaneous ignition source bins (Pumps, Transformers, and Ventilation Subsystems	**Resolved** - Chapter 6; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Fire Propagation from Electrical Cabinets (FAQ 08-0042)	Need to clarify the guidance for addressing the propagation of an electrical cabinet fire beyond the cabinet. Need to clarify those characteristics of cabinets needed to prevent propagation of a fire.	**Resolved** - Chapter 8; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Location of the Fire within an Electrical Cabinet (FAQ 08-0043)	Need to clarify guidance on proper placement of a fire source inside an electrical cabinet when performing fire modeling.	**Resolved** - Chapter 12; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]
Main Feedwater (MFW) Pump Oil Spill Fires (FAQ 08-0044)	Need to clarify the guidance on ignition frequency and severity factor (in terms of oil spill size). Given the large quantity of oil found in a typical MFW pump, even assignment of the "small fire" severity level (i.e., 10% of the oil) leads to postulation of a fire that is actually very severe.	**Resolved** – J. Giitter (NRC) letter to B. Bradley (NEI), June 21, 2012, ML12171A583 [91]
Incipient Fire Detection Systems (FAQ 08-0046)	Need to provide guidance on how to treat / credit a "very early warning smoke detection system" (sometimes referred to as an "incipient fire detection system").	**Interim Guidance** - Chapter 13; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14] **Updated Information** - Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities DELORES-FIRE (draft published: NUREG-2180 [134])
Spurious Operation Probability (FAQ 08-0047)	Need to provide more clear guidance on the determination of circuit failure probabilities (spurious operation probabilities) for components with multiple cables within a fire area.	**Interim Guidance** - Chapter 15; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14] **Updated Information** - NUREG/CR-7150 and EPRI 1026424/3002001989 Vols. 1 and 2 [92,93]
Fire Ignition Frequency (FAQ 08-0048)	Need to update fire ignition frequencies using additional fire event data developed by industry. There appears to be a change in the operating experience and data trends after 1990.	**Interim Guidance** - Chapter 10; EPRI 1019259 and NUREG/CR-6850, Supplement 1 [14] **Updated Information -** EPRI 3002002936 and NUREG-2169 [68]

*The first two columns of this table are taken from Stetkar et al. [19].

Table 9. List of fire PRA FAQs, 2011 (2 of 2)*

FAQ	Issue / Concern	Current Status
Cable Tray Fire Propagation (FAQ 08-0049)	Need to clarify guidance for cable tray to cable tray fire propagation.	**Interim Guidance –** A. Klein, Memorandum to File, July 30, 2009, ML092100274 [95] **Updated Guidance -** NUREG/CR-7010 Vols. 1 and 2 [96,97]
Manual Non-Suppression Probability (FAQ 08-0050)	Need to update guidance for the treatment of manual suppression and the fire brigade response (timing for fire brigade response and the effect on non-suppression probability).	**Interim Guidance -** Chapter 14; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14] **Updated Information -** EPRI 3002002936 and NUREG-2169 [68]
Hot Short Duration (FAQ 08-0051)	Need to provide guidance for the probabilistic treatment of the duration of hot shorts and spurious actuations in AC circuits and the conditions under which this duration is to be applied.	**Interim Guidance -** Chapter 16; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14] **Updated Information -** NUREG/CR-7150 and EPRI 1026424/3002001989 Vols. 1 and 2 [93,94]
Transient Fires – Growth Rates and Control Room Non-suppression (FAQ 08-0052)	Need to clarify and update the guidance on the treatment of transient fires in terms of both manual suppression and time-dependent fire growth modeling (which suppression curve to apply to transient fires in the Control Room and trash can / trash bag transient fires).	**Resolved -** Chapter 16; EPRI 1019259 and NUREG/CR-6850, Supp. 1 [14]

*The first two columns of this table are taken from Stetkar et al. [19].

Table 10. List of NEI fire PRA issues, 2013*

Issue	Concern	Current Status
Hot Short Probabilities	The probability of fire-induced short circuits ("hot shorts") provided in EPRI 1011989/NUREG/CR-6850 is overly conservative for certain configurations.	**Resolved -** NUREG/CR-7150 and EPRI 1026424/3002001989 Vols. 1 and 2 [93,94]
DC Circuit Hot Short Duration	In the absence of experimental data for the duration of fire-induced hot shorts in direct current (DC) circuits, licensees need to assume the short circuits will not clear.	**Resolved -** NUREG/CR-7150 and EPRI 1026424/3002001989 Vols. 1 and 2 [93,94]
Incipient Detection	RES tests for the performance of incipient fire detection systems are not representative of industry applications; licensees are not allowed to take credit for the effectiveness of these systems based on NPP and non-nuclear industry experience.	**Pending -** Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities DELORES-FIRE (draft published: NUREG-2180 [134])
Heat Release Rates (HRR)	RES tests for the HRRs of electrical cabinet fires represent only the upper end (most severe portion) of the range of such fires. Licensees are not allowed to use other sources of data to anchor the lower end of the HRR range.	**Pending -** Draft report for comment: EPRI 3002005578 and NUREG-2178, Vol. 1 [98] Draft report for comment: NUREG/CR-7197 [135]
Mismatch of Fire Data and Fire Scenarios	The data used to estimate the frequency of electrical cabinet fires (which come from operational experience) and the data used to estimate the likelihood of electrical cabinet HRRs (which come from experiments) are inconsistent.	**Pending -** Draft report for comment: EPRI 3002005578 and NUREG-2178, Vol. 1 [98] Draft report for comment: NUREG/CR-7197 [135]

*The second column of this table summarizes the concern as raised by NEI in [20]; it does not indicate the staff position's regarding the concern.

4.4 Fire PRA Applications Environment and Infrastructure

Fire PRA, as with the overall PRA of which it is a part, is performed to support practical decision making.[ff] Some decisions may require information from a detailed, highly realistic PRA; others may require less. Clearly, the PRA's intended application can affect its performance (and realism).

In this section, we briefly discuss, from a U.S. perspective, the fire PRA application environment and the infrastructure (including staff to perform and review analyses, supporting consensus standards and guidance, and training resources) available to support applications.

4.4.1 Fire PRA Applications Environment

As discussed earlier in this paper, fire PRA has long been used to support regulatory decision making. Fire PRA was an integral and important part of the industry-sponsored 1982 Indian Point PRA performed to inform regulatory decisions regarding the continued operation of that plant and the imposition of major backfits to mitigate the effects of a severe accident, and of the NUREG-1150 PRAs (1990), whose results, as discussed in SECY-12-0123 [100], have provided the basis for the CDF and LERF guidelines currently used in numerous regulatory processes. In the mid-late 1990s, fire PRAs were used in the IPEEE program to identify potential vulnerabilities and potentially effective plant upgrades.

In recent years, the focus of U.S. fire PRA activity has been on supporting NFPA 805 transitions, notably the evaluation of changes to previously-approved fire protection program elements.

For the purposes of our discussion on fire PRA realism, the following points are noteworthy.

- NFPA 805, as endorsed in 10 CFR 50.48(c), following the lead of Regulatory Guide (RG) 1.174 [101],[gg] requires that the change evaluation be risk-informed in that it consider risk, defense-in-depth,[hh] and safety margins in an integrated fashion.
- NFPA 805 also requires that:
    - the PRA approach, methods, and data used in the change evaluation be acceptable to the "Authority Having Jurisdiction" (AHJ), i.e., NRC in this case;[ii]
    - the risk impact of plant changes (measured in terms of changes in CDF and LERF) be acceptable to the AHJ;[jj] and
    - when changes are combined into a group for the purpose of evaluation, the impact of individual changes as well as of the combined change shall be evaluated.

[ff] *Recently, one PRA pioneer has expressed a view that current PRA efforts addressing external events (including fire) are "off track" [99]. His concern is that these efforts are focused on answering the wrong question: what is the risk from a particular external event, rather than how that external event affects overall risk.*

[gg] *RG 1.174, which was published in July, 1998 (it was first published as a draft regulatory guide in June, 1997), was an important resource for the NFPA 805 writing group, which started its efforts in Spring, 1998.*

[hh] *NFPA 805 defines defense-in-depth in terms of three barriers: fire prevention; rapid detection, control, and extinction to limit fire damage; and protection of important SSCs to ensure essential plant functions.*

[ii] *The specific NFPA 805 requirement also states the need for the PRA to be appropriate for the change, to be based on current plant conditions, and to reflect plant operating experience.*

[jj] *NFPA 805 references RG 1.174 as a relevant example, but does not, itself, specify criteria.*

- When evaluating an NFPA 805 LAR, the NRC staff addresses the above requirements when making the overall determination as to whether the PRA is of sufficient quality to support the application, i.e., the transition of the fire protection program. (Regarding the requirement for the evaluation of combined changes, the staff follows the guidance of RG 1.205 [102], which, in turn, references the guidance of RG 1.174 in determining which individual changes need to be quantitatively assessed and what changes may be combined together in a group.)
- Depending on the specifics of the LAR, the NRC staff may allow licensees to self-approve fire protection program changes (i.e., to make changes without prior NRC approval) [102].

Thus, just as with other risk-informed applications, the emphasis is on ensuring that the fire PRA is sufficiently realistic (in a risk-informed decision making environment) for the particular application. This emphasis reflected in the NRC staff's Safety Evaluations for the Shearon Harris and Oconee NFPA 805 pilot applications [103,104]. For example, in the Oconee evaluation, the staff's finding that the fire PRA's weaknesses and limitations were "not expected to change the substantial estimated risk decrease associated with this transition into a risk increase" supported its acceptance of the LAR. As another example, in both the Harris and Oconee evaluations, the staff accepted licensee analyses indicating that plant changes resulting in reduced risk from internal events scenarios will offset fire risk increases identified in the change evaluation. This indicated a degree of comparability of the internal events and fire PRA results considered to be adequate for the decision at hand.

We note that the NFPA 805 application, as with most other risk-informed applications, does not require the estimation of pre-core damage endstate frequencies (e.g., the frequency of electrical cabinet fires causing damage to external cables, the frequency of fire-induced LOOP events).[kk] It is, of course, a good modelling practice to compare such estimates against operational experience when relevant data are available. We expect that ongoing efforts to collect and analyze operational experience data (e.g., [51,52]) will provide useful information to support such comparisons.

Looking past the current focus on NFPA 805, we observe that: (a) licensees (and NRC) have invested considerable time and resources to perform (review) fire PRAs, (b) NRC and industry still desire to increase the use of risk information in regulatory applications per the NRC's 1995 PRA Policy Statement [106], and (c) there is growing recognition that many risk-informed efforts may have relied too heavily on the results of internal events analysis [106]. This recognition has been reinforced by the March, 2011 Fukushima Dai-ichi reactor accidents which involved a seismic event coupled with external flooding [108]. Thus, we can expect interest to increase the use of fire PRA in current and new risk-informed applications.

In some cases, these applications may not place any greater demands on the realism of fire PRA. For example, Appendix C shows that for one plant, the ranking of components (as indicated by the risk increase ratio importance measure for basic events) developed using an internal events model is essentially the same as that developed using an all-hazards model that includes fire. In other cases, the potential distortion of a plant's risk profile, as indicated in Figure 8, might need to be investigated and addressed. We emphasize that fire PRA is an iterative process and the realism of a given analysis is a function of the level of effort as well as of the supporting technology. As stated by Stetkar et al. [19],

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[kk] *The evaluation of the reactor pressure vessel through-wall cracking frequency required in the risk-informed, alternate Pressurized Thermal Shock rule, 10 CFR 50.61a [105], is one exception.*

"At some point in this successive refinement process, it is inevitable that decisions are made to terminate the analysis efforts and to retain the quantified risk results. There are myriad factors that affect this decision, including considerations of relative contributions to overall risk, available resources, and expected risk reduction benefits from further refinements to the models, data, and analyses. Therefore, the decision about when to terminate a succession of analysis refinements for a particular plant location is also a very plant-specific and analysis-specific conclusion."

4.4.2 Fire PRA Staffing

Fire PRA, as discussed in this paper, is a sub-discipline of nuclear power plant PRA requiring team expertise on a variety of topics.[ll] In addition to plant and systems analysis (e.g., plant response to upset conditions, operator actions, equipment reliability, probabilistic modeling of systems including uncertainty analysis), these topics include fire phenomenology (e.g., the initiation and growth of fires, the effect of fires on target components), SSC response to fire damage (e.g., fire-induced spurious operations), and fire protection measures (e.g., fire detection and suppression systems, fire brigade makeup and training, and fire barriers).[mm]

Being a sub-discipline, historically there has never been a large number of experienced fire PRA practitioners. For the pre-IPEEE and IPEEE studies, most fire analyses were performed by a handful of organizations (both private contractors and national laboratories) using small teams (generally involving a PRA-oriented lead analyst, supplemented as necessary for such specialized tasks as fire modeling). Following the IPEEEs, there was a drop off in interest in fire PRA as both NRC and industry focused on the use of internal events analyses and results in risk-informed applications. Consequently, when several licensees notified the NRC of their intent to transition to NFPA 805 in mid- to late 2000s, relatively few experienced staff were available to perform and review the fire PRAs. As noted by Stekar et al., "comprehensive and consistently conducted peer reviews are very important to provide assurance of the fire PRA technical quality" and "the number of acknowledged experts who are fully qualified to perform these reviews is extremely limited" [19].

Ongoing training, both formal (as discussed in Section 4.4.4) and informal (through the performance of actual analyses and reviews), is increasing the pool of trained staff, but time is needed to fully reap the benefits of such training.

[ll] *The broad fire protection engineering community also employs risk assessment [109]. Major applications involve the development and support of innovative solutions for situations not covered by existing building codes. Unlike nuclear power plant PRA, much of the focus is on life-safety concerns (i.e., the risk to building occupants), and the target audience of the assessments often are local authorities who must determine whether to accept a proposed, non-traditional, risk-informed solution. Over the years, the fire protection engineering risk assessment community has remained cognizant of the nuclear industry's approach to risk assessment and management and has followed its lead in many ways (e.g., in formulation of risk as a triplet per Kaplan and Garrick [110] and in using risk-informed rather than risk-based approaches).*

[mm] *The earliest fire PRA, performed to address cable spreading room fires in a high-temperature gas reactor design, used a non-phenomenological, statistically-based model to address the likelihood of fire-induced damage to key SSCs [10]. The problem with a statistically-based approach is that not only are the data sparse, they are of uncertain relevance. Fire growth is a highly nonlinear, threshold phenomenon; small differences in physical conditions can lead to dramatic differences in event progression. This characteristic of fire was the underlying reason for the current methodological approach which uses mechanistic fire models to address fire growth and fire-induced damage to SSCs.*

4.4.3 Fire PRA Standards and Guidance

PRA standards (which provide analysis requirements – the "what's") and guidance (which provide the methods – the "how's") can help ensure that a PRA analysis is performed to a level of quality appropriate for a given regulatory application.

In the case of nuclear power plant fire PRA, numerous standards and guidance documents are available (see Figure 9). Regarding standards, in addition to the previously mentioned NFPA 805 standard [12], the currently NRC-endorsed American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard [111] provides a considerable set of requirements for fire PRA. As indicated by NEI [20], several industry fire PRAs have been peer reviewed against these requirements, including the studies that served as the starting point for the three SPAR-AHZ models discussed earlier in this paper.

Regarding guidance, the principal sources of current guidance are the previously mentioned EPRI 1011989/NUREG/CR-6850 [13] and Supplement 1 to NUREG/CR-6850 [14]. RG 1.200 provides high-level guidance for fire PRA consistent with the requirements in the ASME/ANS PRA Standard. Additional detailed guidance on particular topics can be found in NUREG and NUREG/CR reports[nn] and other publicly available documents.[oo]

The existence of relevant consensus standards and guidance and the performance of peer reviews do not, of course, guarantee the realism of a fire PRA. In general, standards and guidance codify current practices, including working solutions when there are recognized gaps in knowledge (the "known unknowns"), but do not necessarily provide final solutions for those gaps. This is illustrated by the concerns discussed in Sections 4.1 and 4.2 of this paper, concerns which have arisen from the results of peer-reviewed fire PRAs.

It is also important to recognize that consensus standards, because they are developed to represent a community viewpoint, will not require detailed analyses of issues currently viewed as unimportant by the technical community.[pp] We discuss a non-fire related example (associated with the 1999 flood at Blayais [113]) in Section 4.4.5, below.

Finally, we note that that current fire PRA standards and guidance are aimed at analyses of CDF and LERF (i.e., Level 1 PRAs) for plants operating at power. Guidance for fire PRA methods addressing non-power operating modes, whether implicit (e.g., in the form of actual studies [114]) or explicit (e.g., [39]) is less well developed. Little work has been done to date on the treatment of fires as part of a Level 2 (post-core damage) PRA, whether considering the possibility of unique post-core damage accident progressions following a fire-induced core damage event or the possibility of non-hydrogen explosion related fires induced by post-core damage accident progression.

[nn] *NUREG, NUREG/CR, and other NRC reports can be readily found using the links on http://www.nrc.gov/reading-rm/doc-collections/#nuregs.*

[oo] *An ADAMS search employing the phrase "Closure of National Fire Protection Association 805 Frequently Asked Question" for Document Type "Memoranda" will identify numerous memoranda (similar to Ref. 95) providing guidance on particular technical issues.*

[pp] *It's worth noting that PRA model reviewers, when presented with a particular model, may not readily identify pieces missing from that model absent external aids (e.g., comprehensive standards and guidance, operational experience, comparable analyses). This is demonstrated by Fischoff, Slovic, and Lichtenstein in their study of the effect of different fault tree model representations [112].*

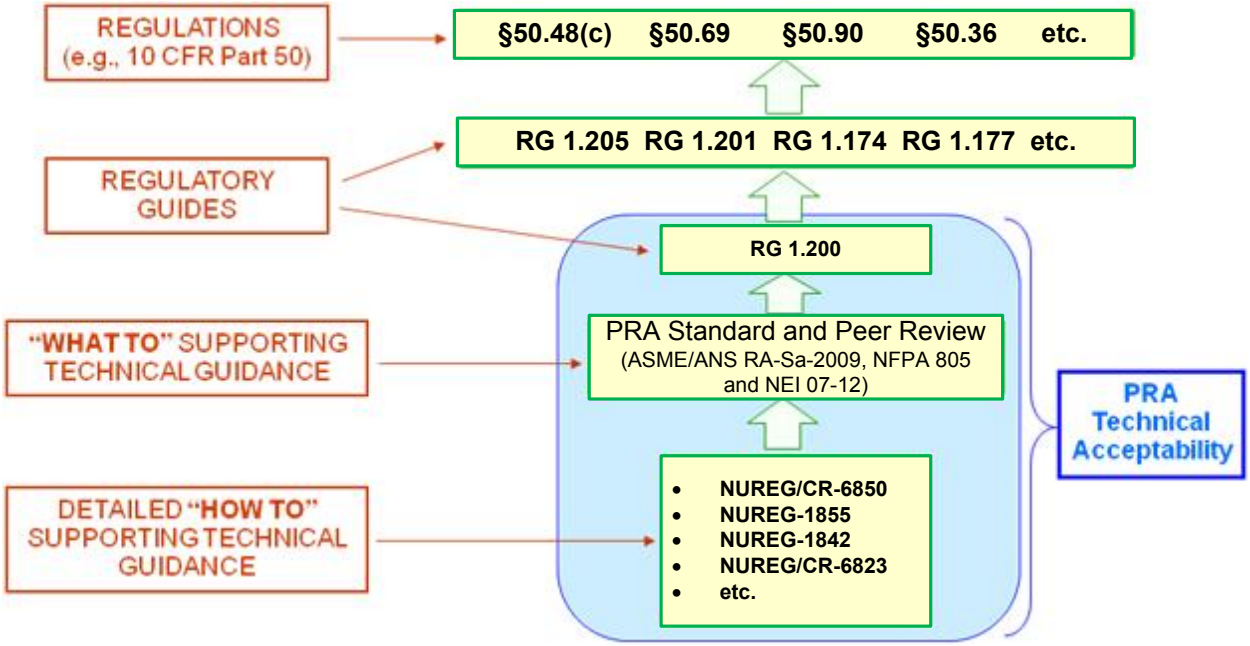

Figure 9. Fire PRA Standards and Guidance Documents

4.4.4 Fire PRA Training

PRA training can help ensure that analysts and reviewers have a consistent understanding of current PRA technology (i.e., PRA methods, models, tools, and data), and can appropriately employ this understanding in performing their work.

In the fire PRA arena, recognizing the breadth and complexity of available fire PRA guidance, NRC and EPRI have jointly held several training workshops covering fire PRA and related topics (circuit analysis, fire modelling, and human reliability analysis). The focus of the workshops has been on the implementation of EPRI 1011989/NUREG/CR-6850 and associated guidance documents (e.g., NUREG-1921 [115], which addresses human reliability analysis for fire PRA). Over the period 2005-2013, a total of 16 workshops have been held. The average workshop attendance has been around 80 students per workshop; most of the students have been from NRC, industry, or overseas organizations. The course training material has been documented via reports (e.g., [116]) and videos.

Since the workshops cover a number of topics in parallel sessions, a number of students have attended multiple workshops to cover all of the topics. We expect that the total number of individuals passing through the EPRI/NRC training is significantly less than 1300 (the product of the number of workshops and the average number of students per workshop). Nevertheless, it can be seen that a sizable number of potential analysts and reviewers have received classroom training in fire PRA.

We caution that historically, there have been surges of interest in fire PRA (notably in the early days of PRA following the Indian Point studies, during the IPEEE program, and now most recently with the implementation of NFPA 805). Absent a strong driving force, it is possible that with the competing demands of different risk-informed applications, the availability and even number of trained analysts and reviewers may dwindle over time.

4.4.5 On Cultural Biases and Conservatism

Despite four decades of experience in performing and reviewing PRAs, and the availability of standards and guidance built on that experience, it is generally recognized within the PRA community that PRA still involves a certain amount of "art" as well as science. Even in the realm of internal events analysis, analysts often need to exercise personal judgment, e.g., when deciding which components should be included in a system model, or when an intermediate result is "good enough" for the purposes of an analysis (so further analysis refinement is no longer necessary). Analyses of internal and external hazards involve further judgments regarding the treatment of large-magnitude events for which the historical record is either uncertain or non-existent. Given the role of judgment, it seems clear that the attitudes and beliefs of both individuals and technical communities can influence the development and use of PRA models.

One non-fire related example of such an influence at a technical community level involves external flooding. In 1999, a storm at the Blayais nuclear power plant (a French four-unit site) led to a wind-related loss of offsite power to Units 2 and 4, followed by a flooding-induced loss of Unit 1 essential service water (Train A) and of the low-head safety injection and containment spray system pumps for Units 1 and 2, as well as flooding of a number of areas of Units 1 and 2 [113].

The Blayais event is now generally acknowledged as an important indicator of the potential importance of external flooding. However, a review of the conference programs for the Probabilistic Safety Assessment and Management (PSAM) and Probabilistic Safety Assessment (PSA) conferences held after Blayais and prior to Fukushima shows considerable interest in the treatment of internal flooding (especially in the 2008-2011 time period), but relatively little activity (and therefore apparent concern) regarding external flooding [117]. Furthermore, the currently endorsed version of the ASME/ANS PRA standard ("Addendum A" [111]) includes a requirement that, following pre-Blayais NRC guidance on the treatment of external floods in the IPEEE program [118], allows the screening of a non-seismic external event if the design basis for that event meets deterministic criteria provided in the NRC's 1975 Standard Review Plan [119]. The ASME/ANS standard also provides the following text prior to its requirements for external flooding PRA: "*These* [external flooding PRA] *approaches, based on a combination of using of the recurrence intervals for the design-basis floods and analyzing the effectiveness of mitigation measure to prevent core damage, have usually shown that the contribution to CDF is insignificant*." The standard reiterates this point later in the same section ("*...external-flooding risks are generally not found to be important contributors to overall risk at nuclear power plants*") and urges analysts to take as much credit as possible for potentially mitigating factors. It is now generally recognized that, based on licensee analyses performed in response to NRC's request for information [120] per Recommendation 2.1 of the NRC's Fukushima Near-Term Task Force [121], the contribution of external floods to total CDF could be, at least for some plants, be very significant [122]. The latest version of the ASME/ANS PRA standard ("Addendum B" [123]) has modified the screening criterion tied to NRC's Standard Review Plan but retains the same tone-setting text mentioned above. We do not know if and how recent external flooding results indicated by Parry and True [122] will be reflected in upcoming versions of the ASME/ANS PRA Standard.

With regard to the potential effects of attitudes and beliefs not tied to specific technical issues, in 2013, the Nuclear Energy Institute (NEI) sent a letter to the Commission noting, among other things that "*Cultural issues and apparent misunderstanding of the intent and approach of PRA are the root"* of a number of impediments in furthering risk-informed decision making [20]. Figure 10 reproduces the list of particular issues raised by NEI in that letter. These issues touch upon fundamental attitudes and beliefs regarding the proper performance and use of a PRA.

As we have neither a strong factual basis nor technical expertise to objectively analyse the attitudes and beliefs of key stakeholders and their effect on PRA realism, we will not discuss the NEI issues point-by-point.[qq] However, we believe it useful to offer the following observations.

- The use of conservative assumptions as an expedient means of focusing analytical resources on the most important issues is well accepted.
- Despite numerous statements regarding the importance of realism in PRAs (see, for example [99,125]), many, if not most, PRA practitioners are comfortable using conservative modelling assumptions, even while recognizing that the analysis results could be conservative. For example:
    - In 1983, at the conclusion of the Indian Point hearings, the ASLB noted the view of an NRC staff expert that "[PRA] modeling approximations are almost always made in the pessimistic direction and hence tend to exaggerate the risk" [36].
    - In 2014, in a small, informal poll of international fire PRA practitioners conducted by one of the authors of this paper,[rr] a majority of the respondents indicated that the use of conservatism is an acceptable approach for dealing with uncertainties (see Appendix E). Many of the respondents indicated that realism is desirable in principle, but conservatism is an important tool for performing a practical analysis. It was recognized that there is a danger that risk insights could be masked if analyses are overly conservative.[ss]
- As illustrated by our external flooding example, the attitudes and beliefs of PRA practitioners can have non-conservative as well as conservative impacts. The impact of conservative assumptions on overall risk estimates can sometimes (but not always) be mitigated by the greater scrutiny typically placed on important scenarios (as noted by the ASLB [36] and by Stetkar et al. [19]). However, in our experience, less attention is typically paid to what are believed to be non-contributing scenarios. Absent an analyst's or reviewer's questioning of the prevailing point of view (and the in-depth pursuit of such questions can face both resource and social challenges), there appears to be little beyond the occurrence of a notable event (e.g., a significant operational event or a major research finding) to prompt re-examination of prevailing beliefs.

[qq] *We note that in 2002, NRC sponsored a study of NRC staff attitudes and perceptions regarding risk-informed regulatory approaches [124]. That report provided a more positive view of the issues shown in Figure 10, indicating that "the majority of respondents felt that there are significant contributions PRA can make to regulatory practices in the reactor program." However, the report also noted that there were "pockets of disagreement" within the staff.*

[rr] *Although they were not asked to characterize themselves, we believe that, based upon the workshop composition, many if not most of the respondents were general PRA analysts involved with fire PRA, rather than fire PRA specialists.*

[ss] *These findings are consistent with our experience regarding the attitudes of a broad range of PRA practitioners. We note that the pragmatic view may be indicative of a less-than-fully integrated, risk-informed approach to decision making, and of the potential value of improved tools to help integrate PRA information with other sources of information.*

Cultural issues and apparent misunderstanding of the intent and approach of PRA are the root of much of the above problems. These continue to exist many years after the PRA policy statement, both in industry and NRC:

1. The generally deterministic mindset of some industry and NRC technical staff can undermine attempts to produce and use a realistic PRA. Some NRC branches, for instance Technical Specifications, have accepted and promoted the use of risk, but this is not consistent, and the appearance of resistance remains within certain technical branches. This could be improved by an internal NRC process that could better infuse risk-informed thinking at the technical staff level.

2. NRC fire testing in particular is biased towards producing very large fires which skew the outcomes and introduce unrealistic and possibly detrimental results with respect to PRA. Use of accelerants, burners and other measures to cause "burnout" create physically different effects than observed in actual plant fire events

3. There are some who believe PRA is just a way to reduce requirements, despite the use of PRA insights to justify new regulations such as ATWS and SBO.

4. Some in the industry believe that compliance is equal to safety. The fact is that there are conditions that involve elevated risk and PRA should not be expected to always show low risk/Green SDP findings. As the PRA Policy statement implies PRA is a double-edged sword. It can show deterministic requirements to be unduly conservative, but it can also identify safety issues, even today.

5. In some cases, lack of understanding of (or the perception of over reliance on) probabilities and uncertainties can lead to a mistrust of the PRA result or the treatment of PRA as an inscrutable black box. This can lead to dismissal of risk insights and non-informed DID expectations.

6. A true, explicit, and predictable consensus process for modeling issues is sorely needed. The current process, despite many attempts at definition, still tends towards deferral to conservative dissenting opinions rather than consensus realistic methods. Both industry and NRC are culpable and need to jointly improve this process.

Figure 10. Excerpt from NEI letter identifying cultural issues [20]

- There are varying viewpoints within the PRA community regarding the meaning of the CDF estimates resulting from a PRA. These viewpoints include the following.
    - The PRA result is an estimate of a real but unknown system property [110].
    - The PRA result is an index, i.e., a useful measure calculated according to certain rules but without fundamental meaning [108,122].

  Both viewpoints accommodate the reality that PRA results are uncertain and conditioned on analysis assumptions and boundary conditions, and both viewpoints can effectively support risk management activities. For the purpose of our current discussion, it appears to us that an important difference is that the former point of view more strongly encourages attempts to improve realism as measured by comparisons with objective data (if available).

- The above observations hold for PRA in general. Considering the specific situation of fire PRA, which involves the intersection of the fire protection engineering community and the NPP PRA community, we offer the following thoughts.
    - Risk concepts have been used in fire protection engineering (largely for insurance purposes) for a very long time.[tt]
    - PRA, as a tool to support risk-informed decision making, is recognized by the fire protection engineering community.[uu] However, such use is not widespread. Watts and Hall state "Probabilistic analysis is not well established in fire protection engineering, where empiricism, heuristics, and, more recently, physics-based modeling are principally used to make decisions" [131].[vv]
    - The NPP fire protection engineering community was not broadly engaged with fire PRA until the IPEEE program in the mid 1990's. Following the IPEEEs, NPP fire protection engineers were heavily involved in the development of NFPA 805, which does not directly require that the fire PRA be conservative, but does require that the fire PRA be "acceptable to the AHJ." Such a requirement, which is similar to NFPA requirements for other models (e.g., fire models), seems straightforward. However, it can be seen that depending on the policies and preferences of the AHJ, this requirement could lead to an intentionally conservative analysis.

As a closing note to our discussion of cultural issues and their potential impact on PRA realism, we observe that changing attitudes and beliefs is hard. As stated by Branch and Olson [133]:

> *"The literature consistently emphasizes, and research demonstrates, that effecting directed behavioral, cognitive, or cultural change in adults and within established organizations is challenging and difficult, requires persistence and energy, and is frequently unsuccessful. Those who go beyond this caveat uniformly emphasize the importance of a sophisticated understanding of "why things are as they are" as an essential starting point in designing effective change strategies. This understanding needs to extend beyond the levels of culture that are readily observable to the deeper levels that create the most powerful forces for perception and behavior and therefore also the most powerful barriers to change. The relevant levels of organizational culture include:*

[tt] *Hall and Nelson [126] note that as early as the late 1600s, varied insurance premiums were used to reflect differences in risk. They particularly note A.F. Dean's development of a premium schedule in the late 1890s that was based on relative risk considerations and whose concepts are still visible in modern practices.*

[uu] *For example, in 1976, the 14th Edition of the NFPA Fire Protection Handbook has a discussion on "risk management" indicates that fire risk managers should evaluate the likelihood of a fire and the extent of its spread [127]. Interestingly, the previous, 1969 edition of the handbook does not refer to risk; rather it discusses "loss prevention," "loss possibilities," and "tolerable losses" [128]. In 1980, the American Society for Testing and Materials (ASTM) held a symposium on fire risk assessment [129]. More recently, NFPA 551 provides a description of fire risk assessment methods that might be used in fire protection applications [130]. The proximity of early activities to the release of WASH-1400 is probably not coincidental; a number of members of the fire protection engineering community were aware of that study.*

[vv] *It should be recognized that some members of the fire protection engineering community are well aware of NRC- and industry-supported efforts to develop and apply fire PRA methods, models, and tools in NPP PRAs. Presentations on these efforts were made at various fire protection workshops and key tools were reviewed by the then National Bureau of Standards (now NIST) [132]. Hall and Nelson indicate that these fire PRA methods are "the most sophisticated fire risk assessment methods available anywhere in the world for any purpose" [126].*

- *artifacts – outward manifestations, symbols, written documents, stated (espoused) values, technology, etc.;*
- *behaviors – the day-to-day behaviors that reveal the values in action, the modes of interaction, and day-to-day practices;*
- *mindsets (ways of thinking) – the basic assumptions, mental models, beliefs, and organizing frameworks;*
- *emotional ground – the mostly unconscious emotional states and needs that drive actions and reactions; and*
- *motivational roots – the aspirations and motivations of the individuals, groups, and leaders who comprise the organization and their alignment or non-alignment with the basic aspirations and purpose of the organization.*

These cultural levels interact in complex ways. Consequently, initiatives to change an organization's culture typically require coordinated intervention at each of these multiple levels."

One can imagine that the difficulties would be even greater for a technical discipline (which involves multiple organizations and individuals). However, such change is not impossible. Interestingly, the fire protection community may provide a relevant example, as demonstrated by its accommodation of physics-based modelling into what was previously a largely empirical field.[ww]

5. Conclusions

Based on the results of our analysis of available information, we come to the following conclusions.

- Fire PRA is in an intermediate-to-late stage of maturity (albeit less mature than internal events PRA). Should there be a desire to accelerate the rate of maturation, a structured consideration of indicators of maturity would likely be useful, and might suggest possible actions in addition to further R&D.
- The quantitative results of current fire PRAs may be conservative. The degree of this potential conservatism is uncertain. Some of the conservatisms observed in actual analyses may arise from practical modeling choices made to reduce analysis effort, and others are likely due to limitations in current fire PRA technology and guidance.
- From a qualitative standpoint, current fire PRAs compare well with operating experience. Most of the important scenarios identified by the fire PRAs appear to have a basis in past precursor events (U.S. and international), and most of the precursor events represent, at a high level, scenarios involving fire sources and induced plant transients typically included in fire PRAs.

[ww] *Per Hall and Nelson [126], early fire models aimed at the characterization of the hot gas layer within a burning compartment were developed in the 1960s. The Harvard Fire Model, the predecessor of current computer-based zone models, was developed in the late 1970s. Efforts to incorporate fire models in building codes and regulations were initiated in Japan in the 1980s. Broader international efforts on performance-based approaches started in the early 1990s.*

- As a potential realism concern,[xx] we note that some fire PRAs identify switchyard/transformer yard fires as being important risk contributors – this result does not seem to be consistent with operational experience. We also note that current fire PRA technology does not address some notable features of a number of precursor events, including multiple fires, multiple hazards, and non-proceduralized recovery actions. At this point, we cannot assess the quantitative impact of addressing these features.
- Work is underway to improve fire PRA methods, models, tools, and data. We fully expect that this work will improve fire PRA realism. We also expect that, similar to the evolution of internal events analyses, continued performance of fire PRA analyses to support practical applications, and the associated review of such analyses, will improve realism.
- Operational experience reviews, such as those discussed in this paper, should be used to help identify and prioritize remaining gaps for improvement efforts. It is particularly important to work on enabling efficient analysis of situations where current, qualitative results (e.g., scenario rankings) appear to be inconsistent with operating experience. Such work may require improvements in the estimation of CCDPs, as well as in the estimation of fire-induced damage.
- Fire PRA realism is affected by the applications environment and infrastructure for performing fire PRA, as well as by the fire PRA technology.
    - The current NFPA 805 transition environment ensures that licensee fire PRAs are sufficiently realistic to support the transition. Other applications may have greater or lesser demands on fire PRA realism.
    - Ongoing training, both formal and informal, is increasing the pool of trained staff, but time is needed to fully reap the benefits of such training. Furthermore, should interest in fire PRA experience a downswing following the implementation of NFPA 805, given the demands of other risk-informed applications, it is possible that the available pool of practitioners could actually shrink.
    - Fire PRA standards and guidance and associated peer reviews can help ensure that a fire PRA is performed to a level of quality appropriate for a given application, but, as evidenced by industry's stated concerns regarding the state of fire PRA, the existence of such standards and guidance do not guarantee realism. We note that EPRI 1011989/NUREG/CR-6850 provides numerous pieces of guidance explicitly aimed at producing a conservative (but not overly conservative) analysis.
    - The use of conservative assumptions in a PRA as an expedient means of focusing analytical resources on the most important issues is well accepted. Furthermore, many PRA practitioners, while recognizing that realism is desirable in principle, see conservatism as an acceptable approach for dealing with uncertainty in practical analyses.
    - The PRA community's attitudes and beliefs regarding particular topics (e.g., the importance of a given hazard) can have non-conservative as well as conservative impacts.

[xx] *Recalling that PRA needs to deal with rare events, we emphasize that in this paper, the concept of realism addresses the degree to which an analysis represents (to the best of the informed community's knowledge) the technical and organizational system relevant to the decision problem at hand.*

- Changing attitudes and beliefs is difficult. Attempts to effect change need to be well-targeted and sustained.

We recognize that a number of our conclusions may appear to be obvious to some in the PRA community. Nevertheless, given the state of controversy within the field, we have found it useful to perform an independent examination of past arguments using information available to the staff. We believe that both quantitative and qualitative comparisons of fire PRA results with operational experience are extremely valuable, and expect that similar comparisons would also be useful for PRA treatments of internal events and other hazards.

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Appendix A – Acronyms and Abbreviations

AC	Alternating Current
ACRS	Advisory Committee on Reactor Safeguards
AHJ	Authority Having Jurisdiction
AHZ	All-Hazard
ANS	American Nuclear Society
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ASP	Accident Sequence Precursor
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DC	Direct Current
EPRI	Electric Power Research Institute
FAQ	Frequently Asked Question
FEDB	Fire Events Database
HRR	Heat Release Rate
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
LAR	License Amendment Request
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MCR	Main Control Room
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
PORV	Power-operated relief valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
R&D	Research and Development
RCP	Reactor Coolant Pump
RES	Office of Nuclear Regulatory Research (U.S. Nuclear Regulatory Commission)
RG	Regulatory Guide
RMIEP	Risk Methods Integration and Evaluation Program
SPAR	Standardized Plant Analysis Risk

Appendix B – Technical Approach for Comparing Precursor- and PRA-Based CDF Estimates

This appendix outlines the technical approach used to compare estimates for the frequency and number of fire-induced core damage events in the U.S. based on: a) conditional core damage probabilities (CCDPs) for U.S. precursor events generated by the NRC's Accident Sequence Precursor (ASP) Program, and b) fire PRA model results. References are provided for readers interested in additional details concerning the technical basis of the approach.

B.1 Frequency of Core Damage Events

Figure 5, reproduced as Figure B1 for convenience, shows precursor- and PRA-based probability density functions (pdfs) for the total U.S. frequency of fire-induced core damage events. The precursor-based pdf was developed through a Bayesian updating process; the PRA-based pdf was developed through somewhat speculative but not unreasonable assumptions regarding the uncertainties in licensee fire CDF estimates, and then developing a probability distribution for the sum of the uncertain CDFs.

The data and approach used to develop these distributions are described below.

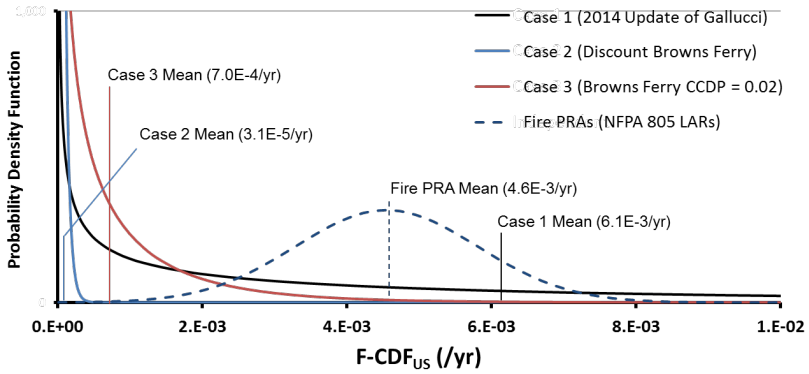

Figure B1. Comparison of precursor- and PRA-based distributions for F-CDF$_{US}$

B.1.1 Precursor-Based Probability Density Function

B.1.1.1 Precursor-Based Fire CDF for an Average Plant

Each of the Case 1-3 curves shown in Figure B1 was developed by developing the distribution for the fire CDF for an "average plant" (conditioned on the data set for each case) and then effectively multiplying this distribution by the number of operating plants to develop the distribution for the total frequency of fire-induced core damage events in the U.S.

In all three cases, the distribution for the fire CDF of an average plant was developed through a Bayesian update of a constrained non-informative prior (CNIP) distribution [B1, B2]. The three cases represent three different approaches for developing the CNIP.

Case 1: The mean value of the CNIP is based on the sum of CCDPs for all events up through 2004, including the Browns Ferry fire. The number of reactor years covers all operating plants up through 2004.

Case 2: The mean value of the CNIP is based on the sum of CCDPs for all events between 1980 and 2004). Similarly, the number of reactor years covers all reactors operating between 1980 and 2004.

Case 3: Same as Case 1, except the Browns Ferry CCDP is set to 0.02 (i.e., slightly lower than the lowest CCDP estimate of 0.03 identified by Gallucci [B3] and one-tenth of the value used in Case 1).

For all three cases, the post-2004 data used to update the prior distribution is the CCDP for the 2010 H.B. Robinson fire (4E-4) and the number of reactor years between 2004 and 2014 (1036). In this analysis, we are not accounting for the reduction in reactor years due to plant outages. The numerical effect of this approximation is minor.

Since the CNIP distribution is of gamma form, the posterior distribution is also a gamma distribution:

$$\pi_1(\lambda|R,T) = \frac{\Gamma(\alpha' + \beta')}{\Gamma(\alpha')\Gamma(\beta')}\lambda^{\alpha'-1}e^{-\beta'\lambda} \qquad \text{(B.1)}$$

where

$\pi_1(\lambda|R,T)$ is the posterior pdf for λ (the fire CDF)
$\Gamma(\bullet)$ is the mathematical gamma function

<u>Evidence</u>:
$R = 4\text{x}10^{-4}$
$T = 1036$ ry

<u>Posterior Distribution Parameters</u>:
$\alpha' = \alpha + R$
$\beta' = \beta + T$

<u>Parameters for a CNIP distribution</u>:
$\alpha = 0.5$
$\beta = \alpha/E_{Prior}[\lambda]$
$E_{Prior}[\lambda]$ = mean value of λ for prior distribution (e.g., $7.1\text{x}10^{-5}$/ry for Case 1)

Table B1 shows the evidence used for the three cases and the resulting values of α and β. Table B2 provides analogous information for three sensitivity cases. These cases involve:

- a direct update of a Jeffreys prior distribution [B1,B2] using evidence of R = 0;
- a direct update of a Jeffreys prior distribution, where R is the sum of all fire-related CCDPs (i.e., the Gallucci estimate plus the CCDP for the Robinson fire); and
- a direct update of a non-informative prior distribution as defined by Winkler [B4] where R is the sum of all fire-related CCDPs.

The first sensitivity case shows the result of a conventional treatment of a situation where no events have been observed. The other two sensitivity cases show the effect of using different diffuse prior distributions.

The mean values and key percentiles of the resulting posterior distributions are shown in Table B3. Table B3 also shows the conventional chi-squared upper bound estimate.

Comparing the results for Precursor Cases 1, 4, and 5 (which are using essentially the same sets of evidence), it can be seen that the updating method used affects the posterior distribution. (This reinforces the message that should an analysis of weak evidence be needed to support decision making, non-informative prior distributions should be used only for screening purposes; for the final results, efforts should be spent on developing an informative prior distribution that appropriately reflects the current state of knowledge [B2].) However, for the purposes of our discussion, it appears that the treatment of Browns Ferry has a greater qualitative impact on the results, and so we do not further pursue these sensitivity cases in this paper.

Figure B2 shows the probability density functions for the Cases 1-3.

Table B1. Evidence for estimating fire CDF and posterior distribution parameters

	R	T (ry)	α	β (ry)
Precursor Case 1, CNIP (mean = 7.1E-5/ry[a])	4E-4	1036	0.5004	8267
Precursor Case 2, CNIP (mean = 3.1E-7/ry[b])	4E-4	1036	0.5004	1.626E6
Precursor Case 3, CNIP (mean = 7.1E-6/ry[c])	4E-4	1036	0.5004	7.106E4

[a]Gallucci mean value
[b]Based on post-1980 experience (R_0 = 7.3E-4, T_0 = 2372 ry)
[c]Based on Browns Ferry CCDP of 0.02 (R_0 = 0.0207, T_0 = 2903 ry)

Table B2. Evidence for estimating fire CDF and posterior distribution parameters – sensitivity cases

	R	T (ry)	α	β (ry)
Zero Events, Jeffreys Prior	0	3939	0.5000	3939
Precursor Case 4, Jeffreys Prior	0.2011	3939	0.7011	3939
Precursor Case 5, Non-informative Prior	0.2011	3939	0.2011	3939

Table B3. Estimates for fire CDF (/ry)

	Mean	5th	50th	95th
Precursor Case 1 (mean = 7.1E-5/ry)	6E-5	2E-7	3E-5	2E-4
Precursor Case 2 (mean = 3.1E-7/ry)	3E-7	1E-9	1E-7	1E-6
Precursor Case 3 (mean = 7.1E-6/ry)	7E-6	3E-8	3E-6	3E-5
Sensitivity Cases				
Zero Events, Jeffreys Prior	1E-4	5E-7	6E-5	5E-4
Precursor Case 4, Jeffreys Prior	2E-4	3E-6	1E-4	6E-4
Precursor Case 5, Non-informative Prior	5E-5	4E-10	5E-6	3E-4
Zero Events, Chi-Squared "Upper Bound"	5E-4			

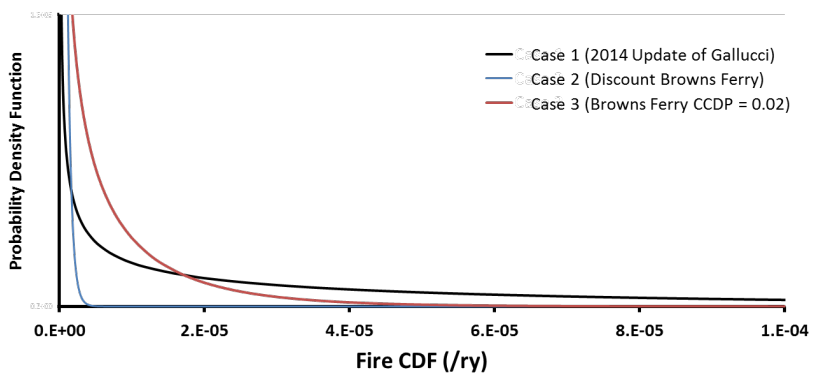

Figure B2. Comparison of fire CDF distributions (/ry)

B.1.1.2 Precursor-Based Frequency of U.S. Fire-Induced Core Damage Events

The frequency of a fire-induced core damage event in the U.S. is the sum of the fire CDFs for U.S. plants:

$$\lambda_{US} = \sum_{k=1}^{N} \lambda_k \tag{B.2}$$

where N is the number of plants. In our calculations, we use a value of 99 for N. Note that in this discussion we use the notation λ_{US} to refer to the term F-CDF$_{US}$ used in Figure B1 and the main body of this paper.

Applying the usual rules for transformation of variables found in probability textbooks (e.g., [B5]), the probability density function for λ_{US} is given by

$$\pi(\lambda_{US}|R,T) = \frac{1}{N}\pi_1\left(\frac{\lambda_{US}}{N}|R,T\right) \tag{B.3}$$

where the function on the right-hand side is the posterior distribution given in Eq. (B.1).

B.1.2 PRA-Based Probability Density Function

The λ_k in Eq. (B.2) are uncertain, but most of the LAR submittals only provide point estimates. To develop a sense of the distribution for λ_{US}, we need to make assumptions about: (1) the marginal distributions for each of the λ_k, and (2) the level of dependency of these distributions.

B.1.2.1 Marginal Distribution for Individual Fire CDF

For the purposes of developing a rough characterization of uncertainty, we assume that each λ_k is lognormally distributed with mean value equal to the LAR point estimate and error factor of 10, i.e., for plant k,

$$\lambda_k = \lambda_{k,LAR} \cdot Z_k \tag{B.4}$$

where $\lambda_{k,LAR}$ is the LAR point estimate and Z_k is a lognormally distributed random variable with parameters $\mu = -0.98$ and $\sigma = 1.40$.

Comparing this assumption with the results of some past fire PRAs (see Table B4[yy]), this assumption seems reasonable. A more accurate analysis would have to account for the shape of the λ_k distributions (they have longer lower tails than implied by a lognormal shape) as well as the plant-to-plant differences.

Table B4. Some Fire CDF Results

Plant	Mean	λ_{05}	λ_{50}	λ_{95}	$\lambda_{95}/\lambda_{50}$	$\lambda_{50}/\lambda_{05}$	$\sqrt{\lambda_{95}/\lambda_{50}}$
Surry 1 [B6]	1.1E-5	5.4E-7	8.3E-6	3.8E-5	4.6	15	8.4
Peach Bottom 2 [B6]	2.0E-5	1.1E-6	1.2E-5	6.4E-5	5.3	11	7.6
Zion (one room) [B7]	1.8E-6	<1E-10	3.3E-8	5.7E-6	170	>330	>240
Indian Point 2 (dominant rooms) [B7]	1.4E-4	5.0E-7	2.8E-5	7.1E-4	25	56	38
Indian Point 3 (dominant rooms) [B7]	9.6E-5	1.5E-6	2.6E-5	9.6E-5	17	3.7	8

B.1.2.2 State-of-Knowledge Dependencies Between Fire CDFs

Lacking licensee information for the uncertainties in the λ_k, let alone their sources, we considered two extreme cases of dependence. For the first case, we assumed the λ_k are epistemically (state-of-knowledge) independent. For the second case, we assumed the λ_k are completely epistemically dependent, such that

$$\lambda_k = \lambda_{k,LAR} \cdot Z \tag{B.5}$$

where Z is a lognormally distributed random variable with parameters $\mu = -0.98$ and $\sigma = 1.40$. In other words, in Eq. (B.4), $Z_1 \equiv Z_2 \equiv \ldots \equiv Z_k \equiv \ldots Z_N \equiv Z$.

[yy] *The significantly larger uncertainties for the Zion analysis are not unexpected; the other analysis results represent the sums of contributions from a number of rooms, and the summation process tends to reduce uncertainties. The large differences between the two Indian Point plants, whose analyses were performed at the same time by the same analysis team using the same methodology, indicates the plant-specific nature of risk.*

B.1.2.3 Distribution of Sum of Fire CDFs

For both dependency cases, $E[\lambda_{US}]$, the mean of the sum in Eq. (B.2) is the sum of the means. Multiplying the average of the LAR fire CDFs by the number of plants (since there are risk-informed LARs for only a subset of all U.S. plants), this value is about 4.55E-3/yr.[zz]

For the case of epistemic independence, the variance of λ_{US} is the sum of the variances for the individual λ_k. We approximate this sum as the product of the number of plants times the average of the LAR fire CDF variances (which is estimated using the assumed lognormal distributions for the λ_k). Recognizing that the sum of a large number of random variables can be reasonably approximated with a normal distribution, it can be shown that the distribution of λ_{US} is approximately normal with parameters μ_I = 4.55E-3/yr and σ_I = 1.26E-3/yr.

For the case of epistemic dependence, Eqs. (B.2) and (B.5) can be used to show that the variance of λ_{US} is given by[aaa]

$$Var[\lambda_{US}] = (E[\lambda_{US}])^2 \cdot Var[Z] \qquad \text{(B.6)}$$

From Eq. (B.5), λ_{US} is lognormally distributed. The parameters of this distribution are: μ_D = -6.37 and σ_D = 1.40.

The results for the two cases are shown in Table B5 and Figure B3. Recognizing the plant-specific nature of risk, and the different states-of-knowledge of the analysis teams performing the various PRAs, we think the epistemically-independent case is probably a better representation of the degree of dependence between the λ_k, and show the results for this case in the main body of this paper.

Table B5. Estimated Distributions for λ_{US}

Case	Mean	5th	50th	95th
Independent	4.55E-3/yr	2.48E-3/yr	4.55E-3/yr	6.62E-3/yr
Completely Dependent	4.55E-3/yr	1.71E-4/yr	1.71E-3/yr	1.71E-2/yr

[zz] *This is the product of the NFPA 805 LAR population mean (4.6E-5/ry) shown in Table 2 of the main body of this paper and the number of plants (99). We provide three-digit precision only to support the traceability of our calculations.*

[aaa] *In general, the variance of the product c·Z for any constant c is the product of c^2 and the variance of Z.*

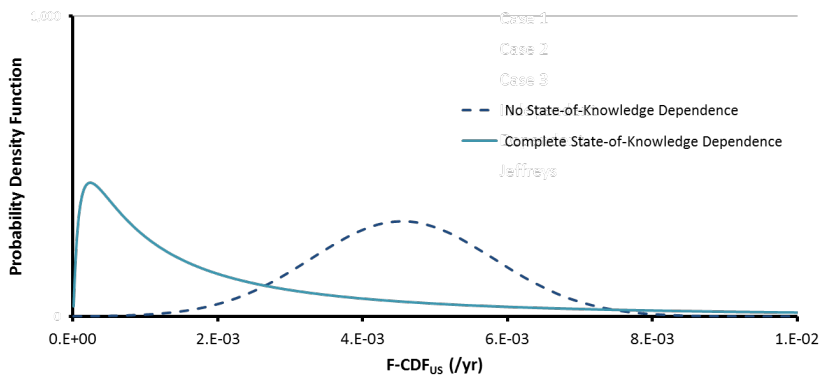

Figure B3. Estimated Distributions for λ_{US}

B.2 Number of Core Damage Events

For a given frequency λ, the number of events occurring in a time period T is implicitly modeled in PRAs as a random variable following the Poisson process.[bbb] The probability of seeing N events in time T is then given by:

$$P(N|\lambda,T) = \frac{(\lambda T)^N}{N!} e^{-\lambda T} \tag{B.7}$$

In our situation, the frequency is unknown. The probability for the number of events is obtained by averaging Eq. (B.7) over the possible values of λ:

$$P(N|T) = \int_0^\infty \frac{(\lambda T)^N}{N!} e^{-\lambda T} \pi(\lambda) d\lambda \tag{B.8}$$

where π(λ) is the pdf for λ.

Table B6 shows the results obtained for two different distributions for λ_{US}: the precursor analysis Case 1 (a Bayesian update of Gallucci's results), and the PRA-based distribution developed assuming that the plant CDFs estimated in the PRAs are independent. Tables B7, B8, and B9 present the results for a number of other cases.

[bbb] *This modeling approximation, which treats event occurrences as the result of a memoryless (non-ageing) process, underlies much of the mathematics built into current PRA models. More complicated probability models to represent time-dependent processes (including burn-in as well as wear-out), are available. These are often used in reliability engineering applications but not, to date, in most PRA-related applications.*

Table B6. Probability of N Events – Precursor Case 1 and PRA Base Case

	Precursor Case 1 (Gallucci Update, T = 10 yr)	PRA Base Case (Independent CDFs, T = 10 yr)
N = 0	0.95	0.96
N = 1	0.051	0.043
N = 2	4.0E-3	1.1E-3
Expected Number*	0.061	0.046

*When λ_{US} is uncertain, the expected number of events is given by $E[\lambda_{US}]\cdot T$.

Table B7. Probability of N Events – Epistemic Uncertainty Sensitivity Case

	PRA Sensitivity Case (Dependent CDFs, T = 10 yr)
N = 0	0.96
N = 1	0.036
N = 2	2.7E-3
Expected Number	0.046

Table B8. Probability of N Events – Time Interval Sensitivity Cases

	Precursor Case 1 (Gallucci Update, T = 50 yr)	PRA Base Case (Independent CDFs, T = 50 yr)
N = 0	0.79	0.80
N = 1	0.15	0.18
N = 2	0.042	0.022
Expected Number	0.30	0.23

Table B9. Probability of N Events – Browns Ferry CCDP Sensitivity Cases

	Precursor Case 2 (post-1980 data, T = 10 yr)	Precursor Case 3 (Browns Ferry CCDP = 0.02, T = 10 yr)
N = 0	~1.00	0.97
N = 1	0.0018	0.031
N = 2	5.0E-6	0.0015
Expected Number	0.0015	0.0015

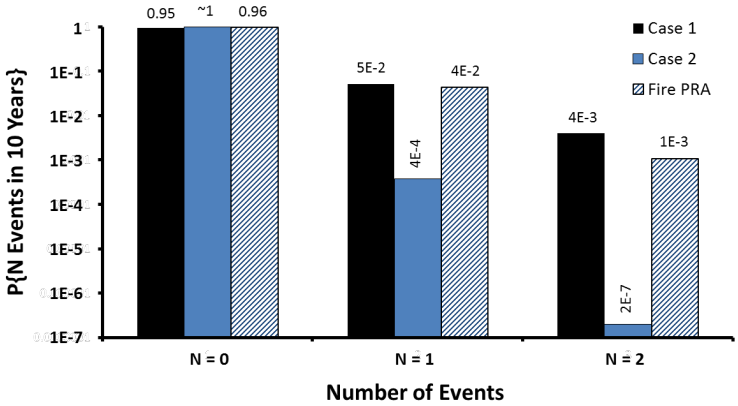

Figure B4. Comparison of precursor- and PRA-based estimates of U.S. fire-induced core damage event probabilities

B.3 References

[B1] Atwood, C.L., et al., "Handbook of Parameter Estimation for Probabilistic Risk Assessment," *NUREG/CR-6823*, 2003.

[B2] Siu, N. and D.L. Kelly, "Bayesian parameter estimation in probabilistic risk assessment," *Reliability Engineering and System Safety*, **62**, 89-116(1998).

[B3] Gallucci, R.H.V., "Predicting fire-induced core damage frequencies – a simple 'sanity check'," *Transactions of 2006 American Nuclear Society Annual Meeting*, Vol. 94, Reno, NV, June 2006.

[B4] Winkler, R.L. and W.L. Hays, *Statistics: Probability, Inference, and Decision, 2nd edition*, Holt, Rinehart and Winston, 1975.

[B5] Papoulis, A., *Probability, Random Variables, and Stochastic Processes*, McGraw-Hill, New York, 1965.

[B6] U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," *NUREG-1150*, 1990.

[B7] Kazarians, M., N. Siu, and G. Apostolakis, "Fire risk analysis for nuclear power plants: methodological developments and applications, *Risk Analysis*, **5**, 33-51 (1985).

Appendix C – Comparison of Basic Event Importance Measures

Table C1 shows the Risk Increase Ratios (RIRs)[ccc] for the most important basic events in a SPAR-AHZ internal events model. Table B2 provides an analogous listing based upon an analysis that includes internal events and internal fires. It can be seen that, at least in this particular case, despite the uncertainties in the model, the basic event rankings are relatively stable.

Table C1. Top basic event importance measures, internal events only

Internal Events Rank	Basic Event Description	Risk Increase Ratio (RIR)*	Basic Event Name
1	CCF 10 OR MORE RCCAS FAIL TO DROP	1.01E+04	RPS-ROD-CF-RCCAS
2	CCF OF RTB-A AND RTB-B (MECHANICAL)	1.65E+03	RPS-BME-CF-RTBAB
3	COMMON CAUSE FAILURE OF AFW PUMPS	1.28E+03	AFW-PMP-CF-ALL
4	CCF 6 BISTABLES IN 3 OF 4 CHANNELS	8.22E+02	RPS-TXX-CF-6OF8
5	CCF 6 ANALOG PROCESS LOGIC MODULES IN 3 OF 4 CHANNELS	8.22E+02	RPS-CCX-CF-6OF8
6	DIVISION 1B 125VDC BUS DP-1B-SB FAILS	7.53E+02	DCP-BDC-LP-1BSB
7	CCF OF ESW MDPs TO START (2)	3.30E+02	ESW-MDP-CF-STRT
8	DIVISION 1A 125VDC BUS DP-1A-SA FAILS	1.76E+02	DCP-BDC-LP-1ASA
9	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO START	1.63E+02	EPS-DGN-CF-STRT
10	DIVISION 1A-SA AC POWER 6.9kV BUS FAILS	1.61E+02	ACP-BAC-LP-1ASA
11	RWST NOT AVAILABLE	1.10E+02	HPI-TNK-VF-RWST
12	CCF – 2 OF 4 EDG E-86 AHUs FAIL TO START	9.72E+01	EPS-FAN-CF-WARMS
13	OPERATOR FAILS TO ESTABLISH CCW FOR CL RECIRC	9.07E+01	RHR-XHE-XM-CCW
14	COMMON CAUSE FAILURE OF RHR MDPs TO START	9.04E+01	RHR-MDP-CF-STRT
	….		
X	CCF OF DG ROOM HVAC FANS TO START	7.19E+01	EPS-FAN-CF-STRT

[ccc] *The Risk Increase Ratio (RIR) for a given basic event is defined as the ratio of: (a) the CDF computed assuming the basic event is failed, to (b) the base case CDF (i.e., the CDF computed using the probability of that basic event). Basic event rankings developed using the RIR are, for many situations of practical interest, nearly identical to those developed using the Birnbaum measure of importance, which is the rate of change of CDF with respect to the basic event failure probability [C1].*

Table C2. Top basic event importance measures, internal events and fire

Internal Events Rank	Basic Event Description	Risk Increase Ratio (RIR)	Basic Event Name
1	CCF 10 OR MORE RCCAS FAIL TO DROP	8.35E+02	RPS-ROD-CF-RCCAS
2	CCF OF RTB-A AND RTB-B (MECHANICAL)	1.35E+02	RPS-BME-CF-RTBAB
7	CCF OF ESW MDPs TO START (2)	1.29E+02	ESW-MDP-CF-STRT
9	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO START	1.15E+02	EPS-DGN-CF-STRT
3	COMMON CAUSE FAILURE OF AFW PUMPS	1.05E+02	AFW-PMP-CF-ALL
12	CCF - 2 OF 4 EDG E-86 AHUs FAIL TO START	7.11E+01	EPS-FAN-CF-WARMS
6	DIVISION 1B 125VDC BUS DP-1B-SB FAILS	6.85E+01	DCP-BDC-LP-1BSB
4	CCF 6 BISTABLES IN 3 OF 4 CHANNELS	6.78E+01	RPS-TXX-CF-6OF8
5	CCF 6 ANALOG PROCESS LOGIC MODULES IN 3 OF 4 CHANNELS	6.78E+01	RPS-CCX-CF-6OF8
x	CCF OF DG ROOM HVAC FANS TO START	5.38E+01	EPS-FAN-CF-STRT
10	DIVISION 1A-SA AC POWER 6.9kV BUS FAILS	2.52E+01	ACP-BAC-LP-1ASA
8	DIVISION 1A 125VDC BUS DP-1A-SA FAILS	2.44E+01	DCP-BDC-LP-1ASA
11	RWST NOT AVAILABLE	1.72E+01	HPI-TNK-VF-RWST
13	OPERATOR FAILS TO ESTABLISH CCW FOR CL RECIRC	1.55E+01	RHR-XHE-XM-CCW
14	COMMON CAUSE FAILURE OF RHR MDPs TO START	1.55E+01	RHR-MDP-CF-STRT

Reference:

[C1] Siu, N. and D.L. Kelly, "On the use of importance measures for prioritizing systems, structures, and components," *Proceedings of 5th International Topical Meeting on Nuclear Thermal Hydraulics, Operations, and Safety (NUTHOS-5)*, Beijing, China, April 14-18, 1997, pp. L.4-1 through L.4-6.

Appendix D – Initial Fire PRA R&D Plans

Table D1 shows potential fire PRA R&D topics identified in 1998 as the result of a structured process considering experiences from the IPEEE program, NRC-sponsored reviews of the then-current state of fire PRA (including available guidance), and feedback from the ACRS. Table D2 shows the fire PRA R&D tasks initiated in FY1998. A number of these tasks were not completed, due to changes in agency priorities (principally in response to the events of September 11, 2001). Numerous reports documenting the results of NRC's fire-protection related R&D activities, including those associated with fire PRA, can be found on the DVD accompanying NUREG/KM-0003 [D1].

Table D1. Potential Fire PRA R&D Topics Identified in 1998 [D2]

Topic	Topic Title	Issue*	Issue Description
T1	Fire events database	I1	Adequacy of fire events database
T2	Fire initiation analysis	I2	Scenario frequencies
		I3	Effect of plant operations, incl. compensatory measures
		I4	Likelihood of severe fires
T3	Fire modeling toolbox: assessment and development	E1	Source fire modeling
		E2	Compartment fire modeling
		E3	Multi-compartment fire modeling
		E4	Smoke generation and transport modeling
		H2	Thermal fragilities
		H3	Smoke fragilities
		H4	Suppressant-related fragilities
		R12	Uncertainty analysis
T4	Fire barrier reliability analysis	B1	Penetration seals
T5	Fire barrier qualification and thermal analysis	B2	Adequacy of data for active and passive barriers
		B3	Barrier performance analysis tools
		B4	Barrier qualification
T6	Detection and suppression analysis	S1	Adequacy of detection time data
		S2	Fire protection system reliability/availability
		S3	Suppression effectiveness (automatic, manual)
		S4	Effect of compensatory measures on suppression
		S5	Scenario-specific detection and suppression analysis
T7	Circuit failure mode and likelihood	H1	Circuit failure mode and likelihood
T8	Impact of fires on operator performance	P3	Fire scenario cognitive impact
		P4	Impact of fire induced environment on operators
		P5	Role of fire brigade in plant response
T9	Risk significance of main control room fires	P1	Circuit interactions
		R1	Main control room fires
T10	Risk significance of turbine building fires	R2	Turbine building fires
T11	Risk significance of containment fires	R3	Containment fires
T12	Fire PRA applications issues	P2	Availability of safe shutdown equipment
		R4	Seismic/fire interactions
		R5	Multiple unit interactions
		R6	Non-power and degraded conditions
		R9	Flammable gas lines
		O3	Comparison of methodologies
T13	Non-core damage issues in fire risk assessment	R7	Decommissioning and decontamination
		R8	Fire-induced non-reactor radiological releases
T14	Precursor analysis methods	R11	Precursor analysis methods
T15	Experience from major fires	O1	Learning from experience
T16	International cooperation	O2	Learning from others
T17	Fire PRA guidance and standardization	O4	Standardization of methods

*Issue code: I = fire initiation, E = fire-induced environment, H = hardware impact, B = fire barrier, S = fire detection and suppression, P = plant response, R = integrated fire risk, O = other.

Table D2. Initial Fire PRA R&D Tasks (FY1998-2000) [D3,D4]

Task	Title
1	Tools for Circuit Failure Mode and Likelihood Analysis
2	Tools for Fire Detection and Suppression Analysis
3	IEEE-383 Rated Cable Fire Frequency Analysis
4	Fire Modeling Toolbox: Input Data and Assessment
5	Experience from Major Fires
6	Industrial Fire Experience
7	Frequency and Characteristics of Switchgear and Transformer Fires
8	Fire Barrier Reliability Model Development and Application
9	Integrated Model and Parameter Uncertainty
10	Frequency of Challenging Fires
11	Fire Model Limitations and Application Guidance
12	Risk Significance of Turbine Building Fires
13	Penetration Seals
14	Multiple Unit Interactions
15	Use of Advanced Fire Models in Fire Risk Assessment

References:

[D1] McGrattan, B., D. Gennardo, T. Pennywell, K. Hill, F.E. Gonzalez, D. Stroup, G. Taylor, N. Melly, and M.H. Salley, "Fire Protection and Fire Research Knowledge Management Digest, 2013," *NUREG/KM-0003*, 2013.

[D2] Siu, N., J.T. Chen, and E. Chelliah, "Research needs in fire risk assessment," *Proceedings of 25th U.S. Nuclear Regulatory Commission Water Reactor Safety Information Meeting*, *NUREG/CP-0162, Vol. 2*, 1997

[D3] Siu, N. and H. Woods, "The U.S. Nuclear Regulatory Commission's fire risk research program - an overview," *Proceedings of International Workshop on Fire Risk Assessment*, *NEA/CSNI/R(99)26*, Nuclear Energy Agency, Paris, France, 2000.

[D4] Siu, N., H. Woods, S. Nowlen, and M. Kazarians, "The U.S. Nuclear Regulatory Commission's fire risk research program: status and results," *Proceedings of PSAM 5, International Conference on Probabilistic Safety Assessment and Management*, Osaka, Japan, November 27-December 1, 2000.

Appendix E – Informal Poll of Fire PRA Experts

In April, 2014, in conjunction with a WGRISK workshop on fire PRA [E1], the workshop participants were asked to anonymously provide written answers to the following questions:

1) Should fire PRA models and results used to support decision making be conservative?
2) When using PRA results in support of decision making, should fire PRA results be added to results for other hazards and initiators (including internal events)?

The full instructions to the respondents are provided in Figure E1.

Fifteen of the 56 workshop participants submitted answers either during the workshop or shortly after. Nearly all of the respondents provided further explanation of their answers, indicating that they saw that the questions as not having simple "yes" or "no" answers.

The survey responses are shown in Table E1. Overall, we note the following.

- The direct answers ("yes's" and "no's") indicate different points of view, but the respondents' explanations indicate that there's significant commonality.
- Regarding Question 1:
  - Nearly all of the respondents provided pragmatic, applications-based viewpoints. (They didn't confine their answer to the principle/philosophy of PRA.)
  - A majority of respondents indicated that the use of conservatism was an acceptable approach for dealing with uncertainty.
  - It was widely recognized that there are large uncertainties in fire PRA models and results and that conservatism is a fact of life. Some respondents discussed how to deal with this conservatism in decision making.
  - The danger of masking important results due to overly conservative analyses was recognized.
- Regarding Question 2:
  - Most respondents indicated that the aggregation of results was acceptable, but also indicated the need to understand the risk contributors.
  - Many indicated that their answer depended on the application (e.g., whether the decision was high-level or focused on plant fire protection).
  - A few respondents indicated that aggregation should not be done because of the differing levels of uncertainty in the analyses of different hazards.

Reference:

[E1] Organization for Economic Cooperation and Development, "Proceedings of OECD/NEA Workshop on Fire PRA, Garching," Germany, Apr. 28-30, 2014, NEA, Paris, France (in preparation).

Table E1. Results of Informal Poll on Fire PRA

	Question 1 (Should fire PRA be conservative?)		Question 2 (Should PRA results be aggregated?)	
No.	**Answer[a]**	**Qualifications[b]**	**Answer[a]**	**Qualifications[b]**
1	Y	Depends on application. Try to be on safe side; don't know magnitude.	Y	Recognizes problems in comparability, advocates comparison of qualitative scenarios (cut sets)
2		Not "should" but a graded approach (with varying degrees of conservatism) is acceptable. Departures from realism need to be justified and considered in decision making.		Answer depends on application. For some decisions (e.g., prioritizing hazards) aggregation is appropriate.
3	N	Conservatism can be tolerated in some applications.	Y	Answer also depends on application.
4	Y	Can't address every scenario in practice.	Y	Answer depends on applications. May not need to aggregate for decisions that depend on dominant scenarios.
5	N	Conservatism is OK for "real uncertainties that cannot be practically removed by further analysis."	Y	Good decision making will always consider contributions.
6		Not "should" but recognizes that analyses have conservatisms to limit analysis resource requirements.	Y	"Of course." Recognizes need for full analyses of all hazards.
7		Not "should" but it "makes sense to be conservative" given uncertainties.	N	Uncertainties are different as compared with internal events. Fire PRA should be used to identify vulnerabilities, suggest improvements.
8	Y	Realistic + uncertainties is the best situation		It depends on the application.
9		Ideally should be realistic but when not possible, better to be conservative.	Y	
10	Y	Be as realistic as possible but use worst possible assumption when the answer is unknown.	Y	Can treat high-uncertainty external hazards separately. Sum internal contributions.
11	Y	Realistic analysis is preferred but it's better to err on the side of conservatism in the face of uncertainty.	Y	Remember that numbers are not as important as relative weights and insights.
12	Y	Aim for realistic, but can use bounding assumptions for non-detailed models	N	There are different levels and kinds of uncertainty.
13	N	Emphatic answer. Realism is preferred, but need to know uncertainty margins.	Y	Emphatic answer.
14	N	Be as realistic as possible.	Y	Model should incorporate all hazards and initiators.
15	N	Conservatism must be transparent, intentional, and applied in a late stage of the process.		"Yes" for high-level decisions, "no" for fire safety solutions.

[a]Y = yes; N = no; blank = direct answer not provided.
[b]Column entries reflect direct or implied qualification to answer.

Informal Poll: Fire PRA Realism

Please provide your personal viewpoint when answering the following two questions. Feel free to provide any explanations, caveats, and/or other comments as you see fit. If you don't have a clear-cut point of view, please indicate this in your answer.

1) Should fire PRA models and results used to support decision making be conservative? (*Optional: please define what you mean by conservative.*)

2) When using PRA results in support of decision making, should fire PRA results be added to results for other hazards and initiators (including internal events)? (*Optional: Why or why not?*)

Figure E1. Instructions – Informal Poll on Fire PRA