PRA RESEARCH AND THE DEVELOPMENT OF RISK-INFORMED REGULATION AT THE U.S. NUCLEAR REGULATORY COMMISSION

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Over the years, probabilistic risk assessment (PRA) research activities conducted at the U.S. Nuclear Regulatory Commission (NRC) have played an essential role in support of the agency's move towards risk-informed regulation. These research activities have provided the technical basis for NRC's regulatory activities in key areas; provided PRA methods, tools, and data enabling the agency to meet future challenges; supported the implementation of NRC's 1995 PRA Policy Statement by assessing key sources of risk; and supported the development of necessary technical and human resources supporting NRC's risk-informed activities.

PRA research aimed at improving the NRC's understanding of risk can positively affect the agency's regulatory activities, as evidenced by three case studies involving research on fire PRA, human reliability analysis (HRA), and pressurized thermal shock (PTS) PRA. These case studies also show that such research can take a considerable amount of time, and that the incorporation of research results into regulatory practice can take even longer. The need for sustained effort and appropriate lead time is an important consideration in the development of a PRA research program aimed at helping the agency address key sources of risk for current and potential future facilities.

KEYWORDS : Probabilistic Risk Assessment; U.S Nuclear Regulatory Commission; Risk-Informed Regulation; Fire Risk; Human Reliability Analysis; Pressurized Thermal Shock

1. INTRODUCTION

Since the completion of the landmark Reactor Safety Study (commonly referred to as WASH-1400) [1] in 1975, probabilistic risk assessment (PRA) results and insights have been used to support regulatory decision making regarding nuclear power plants. Some early, risk-informed[1] regulatory activities undertaken by the U.S. Nuclear Regulatory Commission (NRC) include [2-5]: the prioritization of generic safety issues (1978); the support of licensing hearings for the Indian Point power plant (1983); the review and acceptance of proposed changes to allowed outage times (1980s); and the development of rules relevant to anticipated transients without scram (1984), station blackout events (1988), and plant backfits (1988).

In 1995, supported by two decades of PRA studies (including the NRC's NUREG-1150 study [6] and the licensees' individual plant examinations of internal and external events [7,8]), the Commission issued a policy statement on PRA [9]. This statement directed the NRC staff to increase the use of PRA technology "in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." The statement also addressed the Commission's desires for a reduction of unnecessary conservatism in current regulatory requirements, for realistic PRA evaluations, and for the consideration of uncertainties.

In 1998, as part of its implementation of the PRA Policy Statement, the NRC staff issued Regulatory Guide (RG) 1.174 [10]. This RG provides an acceptable, risk-informed approach that licensees can use to meet NRC requirements regarding changes to a plant's licensing basis. In 2000, the staff revised its reactor oversight program to use PRA models, results, and insights in the planning of inspections, and in the evaluation of inspection findings [11]. In 2004, as discussed later in this paper,

[1]A risk-informed approach to decision making is an approach in which the estimated risk is one of a number of inputs to an integrated decision making process. By contrast, a risk-based approach uses the estimated risk as the sole input.

the NRC developed a risk-informed option for its fire protection rule [12]. Currently, the NRC is using risk information to support the modification of its pressurized thermal shock rule [13], supporting the development of consensus PRA standards and associated guidance [14], and investigating the use of a risk-informed, technology-neutral framework for advanced reactor licensing [15].

The U.S., of course, has not been alone in its development and application of PRA. In fact, as pointed out by Murphy [2], an analytical framework containing the key elements treated by WASH-1400 was discussed by Farmer of the UK Atomic Energy Agency in 1967 [16]. Currently, as evidenced by a recent international survey, PRAs have been performed for most operating nuclear power plants worldwide, and risk information is widely used (in a risk-informed manner) in regulatory and licensee decision making [17].

Throughout this progression towards risk-informed regulation, regulatory research[2] has played a significant role. Starting with WASH-1400 (which assessed the safety of commercial nuclear power plant operation without a specific regulatory application in mind), research has addressed key questions associated with the likelihood and consequences of reactor accidents. These key questions were raised by various sources including PRA reviews (e.g., the Lewis Commission review of the Reactor Safety Study [18]), events (e.g., Browns Ferry, Three Mile Island), risk-informed applications (e.g., the NRC's accident sequence precursor - ASP - evaluation program [19]), and the research community (e.g., researchers interested in the treatment of dynamic effects [20]). With the revival of interest in nuclear power in the U.S., it is expected that research will continue to play an important role at the NRC to address these upcoming challenges (including new risk-informed regulatory applications, new reactor designs, and new technologies).

The purpose of this paper is to discuss the current role of PRA research at the NRC, review selected examples of research activities that led to changes in NRC's regulatory processes, and consider the implications of these examples for future NRC research on PRA and PRA-related topics.

2. PRA RESEARCH AT THE NRC

Broadly speaking, the NRC performs regulatory research to support its mission of licensing and regulating the civilian use of nuclear materials "to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment" [21].

Table 1. Objectives of NRC's Regulatory Research Activities [22,23]

- Ensure that NRC regulations and regulatory processes have sound technical bases.
- Prepare the agency for anticipated changes in the nuclear industry that could have safety, security, or environmental implications.
- Develop improved methods by which the agency can carry out its regulatory responsibilities.
- Develop and maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decision making.

Consistent with this supporting role, the objectives and scope of its various research activities are shaped by the agency's decision making needs. For example, a system designer may be interested in research that identifies an optimal design, while the agency is usually interested in research enabling its review of the acceptability of a range of designs that might be proposed to the agency. In another example, the nuclear industry may perform research to establish a safety case for a proposed change to an existing regulatory requirement, while the NRC might perform research supporting the agency's independent review of the industry research and safety case.

Table 1 provides a set of general objectives for NRC's regulatory research activities identified by the NRC's Advisory Committee on Reactor Safeguards [22] and slightly extended in NRC's recently-developed long-term research plan [23]. Past and ongoing NRC work on PRA and PRA-related topics address a number of these objectives; other objectives suggest potentially fruitful lines of research. Table 2 provides examples of current NRC reactor-related PRA research activities relevant to nuclear power plant applications. The remainder of this section discusses the relationship between these activities and the objectives listed in Table 1.

2.1 Ensuring Sound Technical Bases

NRC's PRA-related regulatory research helps ensure NRC risk-informed regulations and processes have sound technical bases by addressing typical PRA concerns, including completeness,[3] dependencies between modeled

[2] In this paper, the term "regulatory research" refers to the broad range of activities aimed at providing a regulatory agency with new methods, tools, and information to support its decision making.

[3] "Completeness" refers to the PRA's degree of coverage of potentially significant scenarios. Current PRAs, while useful for many applications, have recognized gaps (most notably in the treatment of operator errors of commission) for which there is, as yet, no consensus approach. Completeness concerns can also arise because of the possibility of scenarios of which the PRA technical community is unaware.

Table 2. Examples of Current NRC Nuclear Power Plant PRA Research Activities[1]

Area	Topic	Example Regulatory Research Activity
Reactors	Level 1 internal events at power	Standardized Plant Analysis Risk models (SPAR)
	Level 2	SPAR Level 2, Advanced Level 2/3 methods
	Level 3	Improved MACCS, Advanced Level 2/3 methods
	Low power and shutdown (LPSD)	SPAR LPSD, WGRisk support (2)
	Operational data	Industry Trends Program
	Event analysis	Accident Sequence Precursor (ASP) program, Human Events Repository Analysis (HERA)
	New reactors (evolutionary)	Human reliability analysis (HRA) for new reactors
	Advanced reactors	(3)
	Research and test reactors	
Special Topics	Human reliability analysis (HRA)	HRA empirical study, specific HRA topics (see above)
	Ageing	
	Passive components	
	Passive systems	(3)
	Digital systems	PRA for digital systems (traditional and dynamic)
	Common cause failures (CCF)	International CCF Data Exchange (ICDE) support
	Design and construction	
	Fire	NFPA 805 implementation support, SPAR fire
	Seismic	SPAR external events, updated seismic hazard
	Other external events	WGRisk support
	Security-related events	
	Emergency preparedness & response	
General Systems Analysis Methods and Tools	PRA tools	SAPHIRE (maintenance and development)
	Uncertainty and sensitivity analysis	Treatment of uncertainties in risk-informed decision making
	Advanced computational methods	Binary decision diagrams
	Advanced modeling methods	Dynamic methods (university support, workshops)
	Elicitation methods	
Implementation and Application	PRA quality	PRA consensus standards
	Risk-informed regulation infrastructure	Technology neutral framework (pilot application)
	Risk-informed regulation applications	Pressurized thermal shock rulemaking support
	Risk perception and communication	

Notes:
1) This table identifies areas of current activity. Past NRC work has addressed a number of the gaps shown in the table
2) WGRisk = Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA)/Committee on the Safety of Nuclear Installations (CSNI)/Working Group on Risk Assessment
3) Topic is expected to be addressed under NRC's Advanced Reactor Research program [25]

events,[4] and the accuracy of underlying phenomenological models.[5] The discussions on fire risk and human reliability analysis presented in Section 3 of this paper provide some

[4] "Dependencies" refer to the linkages (both positive and negative) between modeled events due to direct or indirect causes. Examples of such causes include functional relationships between events (e.g., the performance of one piece of equipment depends on the performance of another), common environments (e.g., multiple components are located in the same room), and plant operator decision making during accidents.

[5] In current nuclear power plant PRAs, phenomenological models are used in a number of areas, including the development of success criteria, the analyses of internal hazards (e.g., fire) and external events (e.g., flooding), post-core damage accident progression, and post-release consequence analysis.

examples.

Additionally, a clear understanding of PRA uncertainties and their potential implications for the decision at hand is an important part of the technical basis for a risk-informed decision. Therefore, research activities addressing the characterization, treatment, communication, and management of these uncertainties can be useful. As a particular example, because of their need to address unlikely scenarios, nuclear power plant PRAs must often deal with situations for which empirical data are rare or non-existent. The development of robust approaches to make better use of available information and to characterize the consequent uncertainties in the PRA results is a continuing challenge.

2.2 Preparing for Anticipated Changes in the Nuclear Industry

It is well-recognized that physical changes in existing plants (e.g., software-based control systems) and new design features for prospective plants (e.g., passive safety systems) challenge current, widely-used PRA methods, models, and data. Work is either underway or planned on these topics [24,25]. Other trends that might have implications for PRA research include changes in the regulatory environment associated with the continued implementation of the 1995 PRA Policy Statement, changes to the physical environment, and changes to the political and economic environment. The increased use of PRA technology in regulatory matters can lead to requests for risk results and insights in areas for which PRAs have not been performed. Changes in the physical environment can affect the PRA analyses of extreme weather events. The political changes address the acceptability of new plants which can lead to a need for a large number of new PRAs. Such demand will stress the available resources for performing and reviewing PRAs and could lead to requests for more efficient PRA methods and tools. Economic pressures to reduce plant staffing levels could lead to automation and emergency response concepts which may not be addressed by current PRA technology.

Even trends that don't directly involve the plants could have PRA research implications. For example, ongoing changes in the NRC and industry workforce due to the combination of retirements and new hires will, absent effective mitigative actions, lead to a loss of corporate knowledge regarding the performance and review of PRAs. Research on the best ways to capture, store, and retrieve this knowledge could be a useful adjunct to education and training. Note that such knowledge management work is not solely an information technology issue, as the subject matter expertise of the PRA community is likely needed to support the development of appropriate data structures and search tools.

Workforce changes could also have more subtle effects. For example, a workforce with a higher acceptance of (and expectations for) advanced computation and information technologies might approach the performance, documentation, and use of PRA in a different way. In particular, a workforce trained in the use of computationally-intensive, direct simulation methods for solving complex engineering problems might prefer such methods to the binary logic event tree/fault tree approach, which was developed by the Reactor Safety Study at a time when computational resources were much more limited. Further, such a workforce might expect ready, electronic access to PRA information, even in the field. Research on the needs of such a workforce, meshed with research on computation and information technology trends and their implications for PRA, could help the agency prepare for the future.

2.3 Development of Improved Methods for the Agency

In general, risk-informed regulation represents an improved method for the agency, as it focuses regulatory attention on issues considered to be most important. Research aimed at extending current PRAs to address new issues supports the agency's move to risk-informed regulation. The NRC's work on pressurized thermal shock discussed in Section 3 of this paper provides an example for operating reactors. For advanced reactors, where one potential licensing framework employs frequency-dose curves in the selection of licensing-basis events [15], detailed probabilistic treatments of emergency preparedness and response activities could be useful [26].

PRA research activities aimed at helping NRC's reviewers and decision makers also support the move towards risk-informed regulation. Examples of such activities include efforts to update the SAPHIRE computer program (including new structures and tools aimed at the needs of reviewers) [27], to develop and maintain the Standardized Plant Analysis Risk (SPAR) models used in a number of regulatory applications [28], and to develop the previously-mentioned technology-neutral framework for licensing new reactors [15].

2.4 Infrastructure Development and Maintenance

PRA is a relatively young discipline, and the performance of a PRA study is still something of a art. Thus, on-the-job training is a necessary component of a thorough PRA training program and the performance of PRA research such as the activities mentioned above has an important additional benefit for the agency. In this light, it can be seen that research activities involving the development and application of complete PRA models for specific applications or facilities are likely to be especially beneficial. This benefit is one of the factors considered in the staff's formulation and prioritization of future PRA research projects.

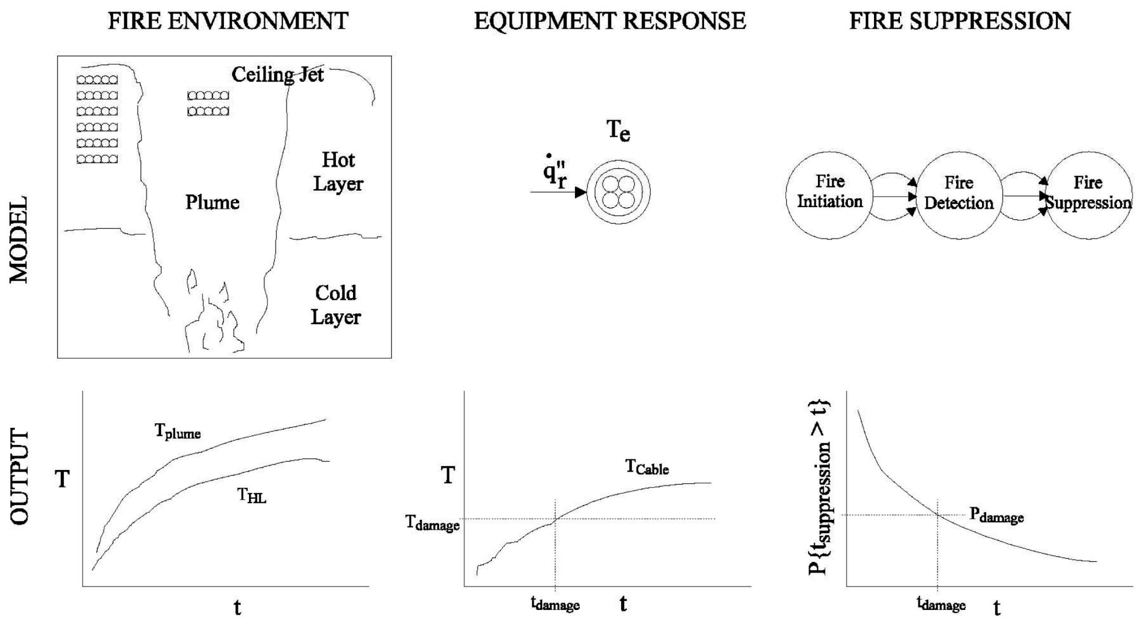

Fig. 1. Elements of Fire PRA Competing Risks Framework [37]

3. NRC PRA RESEARCH AND IMPACT ON RISK-INFORMED REGULATION - EXAMPLES

Section 2 describes the role of PRA research in supporting the NRC's move towards risk-informed regulation. This section provides additional details through the discussion of three examples selected from the experience of one of the authors. These examples involve NRC-sponsored regulatory research in fire PRA, human reliability analysis, and pressurized thermal shock (PTS) analysis.

3.1 Fire PRA Research and Risk-Informed Fire Protection

3.1.1 Background

As described in NRC's Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants" [29], NRC's fire protection requirements prior to the March 22, 1975 fire at the Browns Ferry nuclear power plant [30] were stated in terms of broad performance objectives. These objectives involved: the design and location of systems, structures, and components (SSCs) important to safety; the use of noncombustible and heat-resistant materials; and the provision of fire detection and suppression systems [31]. There was no detailed implementation guidance for determining whether a plant's fire protection program met these objectives and the staff relied upon compliance with local fire codes and insurance underwriter ratings to determine acceptability.

Following the Browns Ferry fire, during which multiple safety systems were lost, the NRC initiated the development of detailed fire protection program requirements. This culminated in the 1980 publication of a fire protection rule (10 CFR 50.48 [12]) and an associated appendix (Appendix R to 10 CFR 50 [32]). Among other things, these documents provided prescriptive, deterministic requirements aimed at ensuring that a plant could achieve and maintain a safe and stable condition following a fire.

From a PRA perspective, the Browns Ferry fire prompted the NRC to perform a limited study of the risk significance of that event. This study was published as a supplement to WASH-1400.[6] It indicated that the core damage frequency (CDF) associated with that fire was around 10^{-5}/yr, or about 20% of the CDF due to causes addressed in the main body of WASH-1400 (e.g., loss of coolant accidents, plant transients). The study also noted the usefulness of developing a more detailed fire PRA methodology (including improved models and data).

3.1.2 NRC-Sponsored Fire PRA Research

In 1977, the NRC initiated a new 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

[6] Note that fires were not addressed in the draft version of the Reactor Safety Study report that was issued for public comment in 1974.

National Laboratories and Brookhaven National Laboratory. The UCLA project was later supplemented by additional NRC-sponsored efforts at the Rensselaer Polytechnic Institute.

Apostolakis and his team developed and demonstrated an approach that integrated the predictions of deterministic models for fire behavior ("fire models") into the assessment [33]. The integration was accomplished through the use of a competing risks framework, in which the probability of fire damage to equipment (including electrical cables) was computed as the outcome of a "race" between two simultaneous processes: fire growth and fire suppression. Figure 1 shows three major elements of this framework: (1) the use of an appropriate fire model to predict the environment in the area of interest; (2) the use of an appropriate heat transfer model (possibly integrated within the fire model) to predict the time to damage for a target (or set of targets) in that area; and (3) the use of a probabilistic model to develop the distribution of the time to suppress the fire. The framework explicitly addressed uncertainties in the predictions of the physical models, as well as uncertainties in the parameters of the suppression model.

The UCLA approach was used in the industry-sponsored Zion (1981) and Indian Point (1982) PRA studies [34,35], which showed that fire could be an important contributor to CDF and risk due to its potential to act as a failure mechanism affecting multiple trains of equipment. The approach was documented in numerous papers and reports, including the 1983 PRA Procedures Guide [36], and provided the basic framework for most subsequent fire PRAs in the U.S., whether performed by industry or the NRC [37].

In the late 1980s and early 1990s, NRC PRAs (e.g., the NUREG-1150 studies [6]) and industry studies (including the Individual Plant Examinations of External Events [8]) 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, [38].) In 1998, the NRC restarted its fire research program (which had been stopped in 1987) in order to address a number of important issues, including the likelihood of fire-induced circuit failures (including spurious actuations), the treatment of fire detection and suppression, and the identification of insights from significant nuclear power plant fire events [39]. In 2001, the NRC's Office of Nuclear Regulatory Research (NRC-RES) and the Electric Power Research Institute (EPRI) initiated a joint effort aimed at developing a basis for fire PRA guidance.

In recent years, industry and NRC have focused their attention on implementing the results of NRC and EPRI research. This has led to NRC's development of a risk-informed option to its fire protection rule (10 CFR 50.48 [12]), support of an associated consensus standard for risk-informed fire protection programs, and participation in the development of related fire PRA standards and guidance, as discussed in Section 3.1.4 below. NRC's fire-related regulatory research has focused on addressing implementation-related topics, including the likelihood of fire events that have the potential to develop into important fire scenarios, the likelihood of various fire-induced cable failure modes and associated circuit faults, and the verification and validation of fire models used in fire PRAs. (The last topic was addressed as part of a cooperative research project with EPRI.)

3.1.3 Technical Accomplishments

Although the NRC-sponsored effort at UCLA did not lead to the first published approach for addressing nuclear power plant fire risk,[7] it did create the analytical framework used in most US fire PRA studies.[8] Unlike other probabilistic approaches for addressing fire (e.g., [40,42]), this framework specifically deals with the strong dependence of fire dynamics on key scenario characteristics, including fuel type, fuel bed geometry, target location, compartment geometry, and ventilation. (This dependence limits the usefulness of generic statistics, e.g., for the size of cable tray fires, developed from operational experience.) The framework addresses this dependence by integrating fire models into the PRA. It can be viewed as an attempt to blend relevant statistics (e.g., regarding fire ignition) with other forms of evidence (e.g., model predictions).

The UCLA approach also directly addresses the question of uncertainties. It differentiates between aleatory uncertainties (also referred to as "stochastic" or "random" uncertainties) and epistemic ("state of knowledge") uncertainties [43], and quantifies the contributions due to uncertainties in model parameters and those due to model structure [44].

NRC's fire PRA research has resulted in more than just an approach. Early work provided tools (including the COMPBRN computer code used to model compartment fires [45]), estimates for key parameters (including fire frequencies [46], electrical cable fire-related properties [47], and parameters characterizing the aleatory distributions for detection and suppression times [48]), and analyzed industry data for fire events [49].

[7] In addition to the WASH-1400 analysis of the Browns Ferry fire, a probabilistic damage-zone oriented approach was developed for a risk assessment of cable spreading room fires in high temperature gas cooled reactors [40].

[8] A number of non-US fire PRA studies [41] have used the "Berry method," named after the lead developer of a method for fire hazards analysis [42]. This method includes, as one of its elements, a modified decision tree method for probabilistic analysis of fire scenarios that was first proposed in 1976. As implemented by Berry, the method is used to determine whether supplementary fire protection measures (e.g., detectors, sprinklers) will meet a plant's fire safety objectives.

Later work applied these fire PRA methods, tools, and data to develop insights for a number of plants (e.g., [8]) and to help assess the importance of specific issues (e.g., the effect of fire protection system actuations on safety-related equipment [50]). Recent work has provided guidance on the treatment of a number of detailed issues (e.g., pump oil spills) [51], an assessment of the importance of a number of fire scenarios involving fire-induced short circuits and supporting empirical data for the behavior of electrical cables in fire environments [52], and a characterization of the strengths and weaknesses of a number of fire models based on a comparison of model predictions to experimental measurements [53].

3.1.4 Regulatory Impact

In 2004, the NRC amended its fire protection rule (10 CFR 50.48 [12]) to add a risk-informed, performance-based option. The amended rule allows licensees to maintain a fire protection program that complies with the National Fire Protection Association (NFPA) Standard 805 developed in 2002 [54], with some specific exceptions described in the rule. Because the risk-informed, performance-based option provides a potentially more efficient means for a plant to comply with NRC's fire protection requirements, a number of licensees are performing, or planning to perform, work needed to support their transition to a risk-informed, performance-based fire protection program as allowed by the amended rule [55]. It is expected that the consensus fire PRA standard developed under the auspices of the American Nuclear Society (ANS) [56] and fire PRA guidance jointly developed by EPRI and NRC-RES [57] will provide useful support to these efforts.

Fire PRA methods, tools, data, results, and insights are also being used by the NRC to focus fire protection inspections (using, for example, information on the likelihood of fire-induced circuit faults [58]) and to assess the significance of findings [59]. Fire PRA is being used by licensees to address fire issues beyond fire protection systems and features (e.g., equipment such as emergency diesel generators) in support of analyses for risk-informed technical specifications.

NRC's fire PRA research, complemented by industry and international efforts, has provided crucial support for these regulatory advances. Comparing with the objectives listed in Table 1, fire PRA research has: (1) provided the technical basis for risk-informed fire protection by enabling the assessment of fire risk; (2) helped prepare the agency for the future by developing tools that could be used in the licensing of new plants; (3) supported the development of improved agency methods for dealing with fire protection issues (including the possibility of fire-induced spurious actuations); and (4) developed computational and human resources needed to support a variety of fire-protection related activities. As a particular example of the last point, project personnel with strong experience in electrical engineering but little prior fire or risk experience have gained enough experience to support to NRC's fire inspections of key circuits.

3.2 Human Reliability Analysis Research

3.2.1 Background

In 1975, WASH-1400 demonstrated that the behavior of plant operators and other staff could be practically addressed in the framework of a PRA study. WASH-1400 also showed that human errors could be important contributors to risk. For example, the study identified human error as the source of roughly 20% of the unavailability of the high pressure injection system of the pressurized water reactor (PWR) studied [1]. (The unavailability of this system was an important factor in a number of the dominant PWR accident sequences.) However, given the controversy surrounding WASH-1400 at the time [2,3], little regulatory use was made of this or other study findings.

The March 28, 1979 accident at Three Mile Island (TMI) Unit 2 changed the agency's view on PRA.[9] It also changed the agency's regulatory approach to human factors, as described in the agency's 1980 action plan [60]. Personnel activity in the control room was addressed in 1981-1982 through NRC-mandated licensee reviews of the human factors interface with three areas: control room design, safety parameter display systems, and emergency operator procedures. The requirements for each of these areas were codified in 10 CFR 50.34(f) [61] and in updates to NRC's general design criteria [31].

In conjunction with its new regulations governing control room activity, the NRC also addressed issues about personnel staffing, qualification, and licensing [62-66]. NRC's expectations about operator qualifications and training were later clarified in a 1986 policy statement on engineering expertise [67] and were later enhanced to reflect the value of an engineering degree in a 1989 policy statement [68].

None of these post-TMI policy or rule changes regarding human factors were risk-informed. However, the NRC's post-TMI plans regarding human factors did include activities to include human factors data into PRA through human reliability analysis (HRA). Moreover, the earlier Lewis Commission's review of WASH-1400 had also raised concerns with that study's HRA [18]: (1) the availability of relevant HRA data, (2) inherent methodological limitations in the treatment of time windows, (3) the lack of treatment of potential errors of commission, (4) the lack of treatment of potential recovery actions, and (5) the consequent uncertainties in

[9] Arguably, the general sequence of events that occurred at TMI was captured in a number of the dominant sequences identified by WASH-1400.

the analysis. In parallel with the post-TMI changes, therefore, the NRC sponsored a considerable amount of HRA work, as discussed below.

3.2.2 NRC-Sponsored HRA Research

In general, HRA involves the identification, modeling, and quantification of human failure events (HFEs). HFEs are the basic events in the PRA's event tree/fault tree models that provide the portal through which human factors considerations are brought into the PRA [69, 70].[10]

As described in Volume III of WASH-1400 [1] and Swain [72], the Reactor Safety Study employed the Technique for Human Error Rate Prediction (THERP) method, developed at Sandia National Laboratories in the 1960s in work sponsored by the NRC's predecessor, the U.S. Atomic Energy Commission (AEC). This method involved the decomposition of operator (and other plant staff) tasks into subtasks, the assessment of relevant performance shaping factors (PSFs) affecting those subtasks, and the estimation of the likelihood of subtask failures using a variety of sources of information (e.g., performance data from military aircraft crews, international operational experience, operating crew interviews, and observations of crew performance).

Many of the HFEs addressed in WASH-1400 involved maintenance-related tasks (e.g., sensor calibration, valve re-alignment following maintenance); however, the analysis also treated some key HFEs associated with operator actions during an accident (e.g., the failure of operators to properly initiate the recirculation cooling mode of emergency core cooling during a loss of coolant accident). The HFEs were identified by members of the PRA team responsible for developing system fault trees while the role of the human factors specialists was to quantify the probability of these HFEs.

Following the re-awakening of interest in PRA after TMI, the NRC sponsored a number of HRA methods development activities. At the time, there was a debate in the broader PRA community regarding the best representation of accident-progression related dependencies in an event sequence model. This debate was typically framed in terms of the merits of "large event trees/small fault trees" versus those of "small event trees/large fault trees." The questions regarding HRA were: 1) whether HFEs should be modeled as top events in the event trees (increasing the size of the trees), or whether they should be incorporated as basic events in the fault trees (as was done in WASH-1400); and 2) the appropriate role of HRA analysts (and, more broadly, human factors specialists) in the development of the overall PRA model. Different PRAs took different approaches to these questions, and NRC's HRA development activities should be viewed in this context.

The NRC's HRA development activities shortly after TMI focused on the refinement and documentation of a number of existing methods, including THERP [73], the Operator Action Tree (OAT) method [74], and the Success-Likelihood Index Method (SLIM) [75]. The OAT method is an event-tree based approach that directly addresses operator decision making (and the possibility of mistakes), and accounts for the possibility that operators can, with time, recover from mistakes. SLIM is a structured quantification approach that uses expert judgment to identify key PSFs, weight their importance, and adjust a nominal human error probability (HEP) quantified using data or another HRA method. In this time period, the NRC also sponsored the development of the Maintenance Personnel Performance Simulation (MAPPS) model [76]. Although MAPPS was not used in HRA, its stochastic, task-oriented approach to analysis provided some features echoed in later work.

These methods addressed some of the concerns raised by TMI and the Lewis Commission. However, they did not address a fundamental issue raised by TMI – the possibility that, in addition to failing to accomplish a necessary action (as modeled in the PRA), operators could also take a wrong action, e.g., throttle a needed cooling system (which happened at TMI). In principle, a PRA model could have included such "errors of commission," but there was no basis for quantifying their probabilities. Moreover, there was no formal method for addressing the cause of these errors, which could affect the failure probabilities of subsequent actions (including recovery actions).

In the late-1980s, NRC launched a number of efforts aimed at addressing errors of commission and the broader issue of incorporating causal mechanisms into HRA. The ATHEANA project [71,77,78], perhaps best known of these efforts, emphasizes the identification of error forcing contexts, which are sets of factors (typically scenario-related) judged to strongly increase the likelihood of operator error. Figure 2 illustrates the ATHEANA process. ATHEANA was used in NRC's assessment of PTS risk, discussed in Section 3.3 of this paper.

The ATHEANA approach was designed to directly support the event tree/fault tree models used in current PRAs. A concurrent NRC-sponsored project developed a dynamic PRA modeling framework intended to provide a more detailed representation of the context for operator actions [79]. This framework employed dynamic event trees modeling the time-dependent evolution of different possible plant and crew states (including "diagnosis," "planning," and "quality" states).

[10] The term "human failure event," used by the ATHEANA (A Technique for Human Event ANAlysis) method [71], is used instead of "human error" to both emphasize the connection with the overall PRA model, to ensure appropriate generality (although common usage of the term "error" can connote incorrect action, many HFEs involve the operators' failure to perform necessary actions), and to reduce the unnecessary attribution of blame (since the HFE might involve personnel actions that are in accordance with current procedures and training).

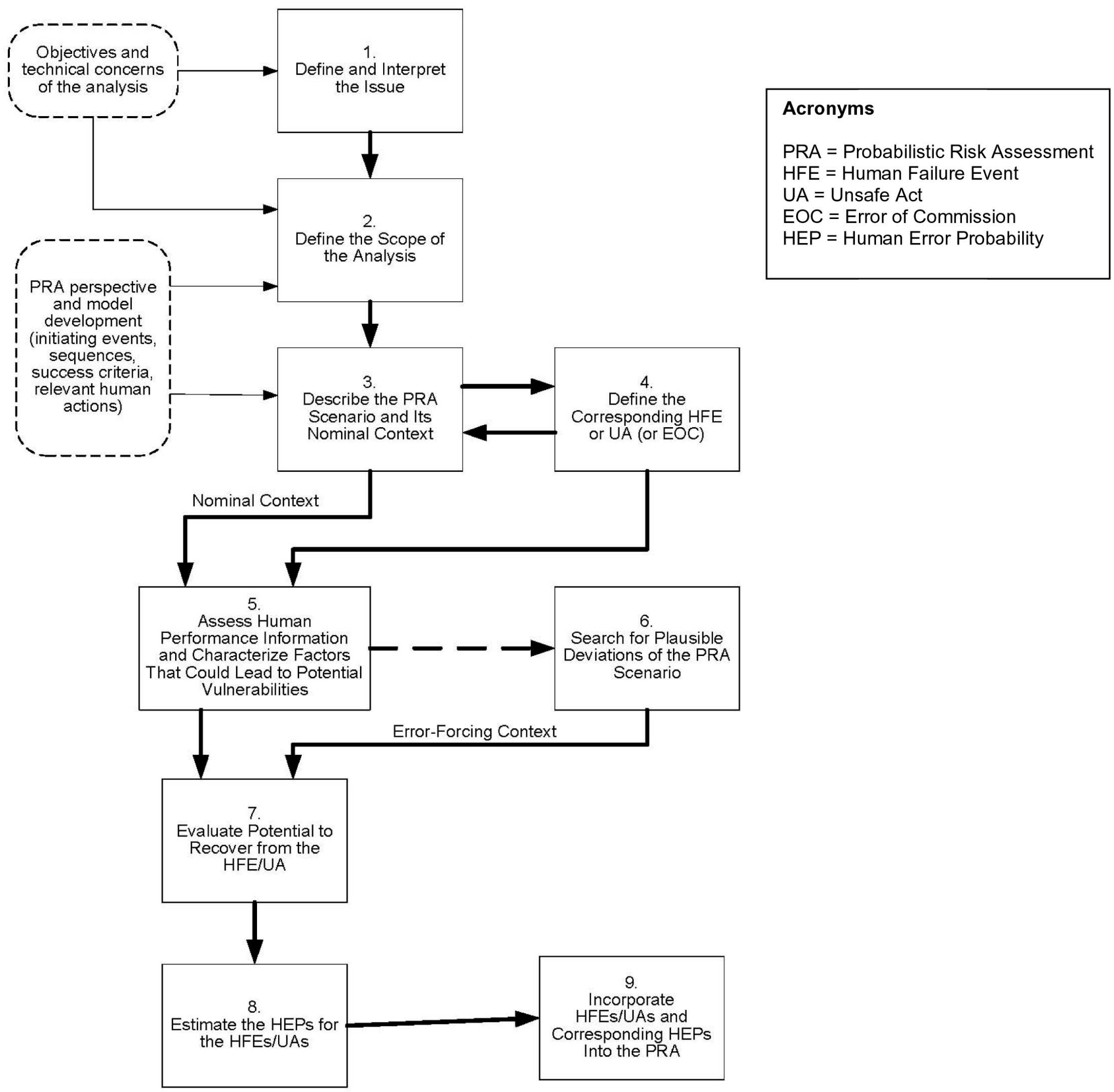

Fig. 2. Steps in ATHEANA Methodology [70]

NRC also sponsored the development of the Cognitive Environment Simulation (CES), a computer model simulating operator intention formation during accident sequences [80] and a smaller-scale effort to develop an object-oriented simulation model of an operating crew [81]. Both of these simulation efforts were deterministic and not incorporated into a PRA. However, they provided insights that may be valuable for current HRA development efforts (e.g., [82]).

Despite the significant amount of effort expended to date, the errors of commission problem has proven to be an extremely difficult one to solve. U.S. efforts (e.g., the ATHEANA project) and parallel, international efforts [83] provide potential approaches, but routine HRA practice does not incorporate any of these approaches. The obstacles to mainstreaming include the large amount of resources needed to employ the methods (a consequence of model complexity), a lack of consensus in the HRA community regarding the detailed contextual elements and mechanisms to be addressed by the methods, and lack of high quality empirical data to calibrate (let alone validate) the models.

With the rise in demand for support of risk-informed applications, NRC's more recent efforts have focused on developing tools (including a simplified HRA method to support NRC's Standardized Plant Analysis Risk – SPAR – models [84], and a human error database [85]) and HRA guidance [86]. NRC has also taken a leading role in the organization and support of the HRA Empirical Study [87], an international project benchmarking HRA model predictions against operating crew performance data developed by the OECD/NEA Halden Reactor Project. This project is expected to identify the strengths and weaknesses of HRA models currently used (or available for use) in nuclear power plant PRAs.

3.2.3 Technical Accomplishments

It is easy in today's PRA environment, where the lack of treatment of human error would be viewed as a serious omission, to take the availability of accepted HRA methods for granted. In fact, as evidenced by vigorous debates over the years over such basic issues as the definitions of "human error" and "human reliability analysis" (e.g., [88-90]), this state of affairs was not easily attained.[11] With this perspective, it can be seen that AEC's and NRC's HRA research activities have resulted in significant technical accomplishments.

Regarding current HRA, AEC- and NRC-sponsored efforts have produced a number of HRA products, including frameworks to perform HRA [91], HRA methods (notably THERP, SLIM, and their variants [84,92,93]), and databases (e.g., [94]). Many of the HRA methods have been widely used in the U.S. and abroad.

Regarding HRA developments, NRC's work on ATHEANA and its support of dynamic PRA developments have contributed to the movement of the HRA field towards so-called second generation methods, i.e., methods that explicitly account for human cognitive behavior and the contextual factors that affect this behavior [70]. NRC's work on HERA will provide an information base that should be a useful repository for many sources of evidence, including simulator experiments as well as operational experience. The ongoing HRA Empirical Study mentioned in the previous section will be a landmark effort, as it will provide the first ever international comparison of HRA model predictions against empirical data.

3.2.4 Regulatory Impact

Unlike fire PRA, NRC's HRA research has not directly led to specific changes in NRC's regulations. However, because of the importance of human error to risk, this research has, by supporting the development of HRA methods and guidance, been a key element in the agency's move toward risk-informed regulation.

In particular, NRC's HRA research efforts have addressed each of the regulatory research objectives identified in Table 1. These efforts have: (1) provided a critical element of the technical basis for current PRAs; (2) supported the development of more phenomenologically-oriented HRA approaches which should be extendable to the analysis of new situations (e.g., advanced control rooms); (3) supported improvements in NRC's Accident Sequence Precursor program and Reactor Oversight Program through the development of appropriate HRA tools; and (4) provided the technical basis for HRA guidance documents and input to PRA consensus standards.

3.3 Pressurized Thermal Shock PRA Research

3.3.1 Background

The PTS Rule (10 CFR 50.61 [95]) provides NRC's requirements limiting radiation-induced embrittlement of PWR reactor pressure vessels (RPV). This rule requires that PWR licensees perform a deterministic evaluation of their RPV's fracture toughness transition temperature at the RPV's end of life. If the computed temperature (denoted as RT_{PTS}) exceeds the screening criteria established in 10 CFR 50.61, the licensee is directed to take steps to reduce the neutron flux to the RPV wall. Plants for which the computed RT_{PTS} value, even with neutron flux reduction, will still exceed the screening criteria are required to submit a plant-specific safety analysis identifying what, if any, plant modifications are needed.

RG 1.154 [96], the regulatory guide supporting the PTS Rule, describes one acceptable method for performing such a safety analysis. This guide calls for an estimate of the RPV through-wall crack frequency (TWCF) and a comparison of the estimated TWCF with an acceptance criterion of 5×10^{-6}/yr.

The technical basis for the PTS Rule (which was originally issued in 1991), was provided in a 1982 Commission Paper (SECY-82-465 [97]). In the years following, NRC- and industry-sponsored research on the materials behavior of pressure vessels provided a means for addressing a number of intentional but unquantified conservatisms in SECY-82-465. In 1999, the NRC initiated its PTS Reevaluation Project, which was aimed at developing the technical basis for a risk-informed revision of the PTS Rule.

3.3.2 PTS PRA Research

The PTS Reevaluation Project, described in NUREG-1806 [13], involved detailed PRA analyses of three PWRs. The analyses addressed internal initiators (e.g., loss of coolant accidents, reactor trips) occurring during both power and shutdown operation. A scoping-level assessment was performed to address the potential impact of internal hazards (e.g., fire, flood) and external events (e.g., earthquakes), and sensitivity analyses were performed to generalize the PTS study results to other PWRs. Figure 3, adapted from NUREG-1806, summarizes the approach used for the detailed analyses.

As shown in Figure 3, a PRA event sequence analysis was performed to define and quantify sequences of events that could, through a combination of system

[11] Interestingly, current concerns regarding software-based systems and their treatment in PRAs involve very similar issues [22]. Significant progress in this area awaits their resolution.

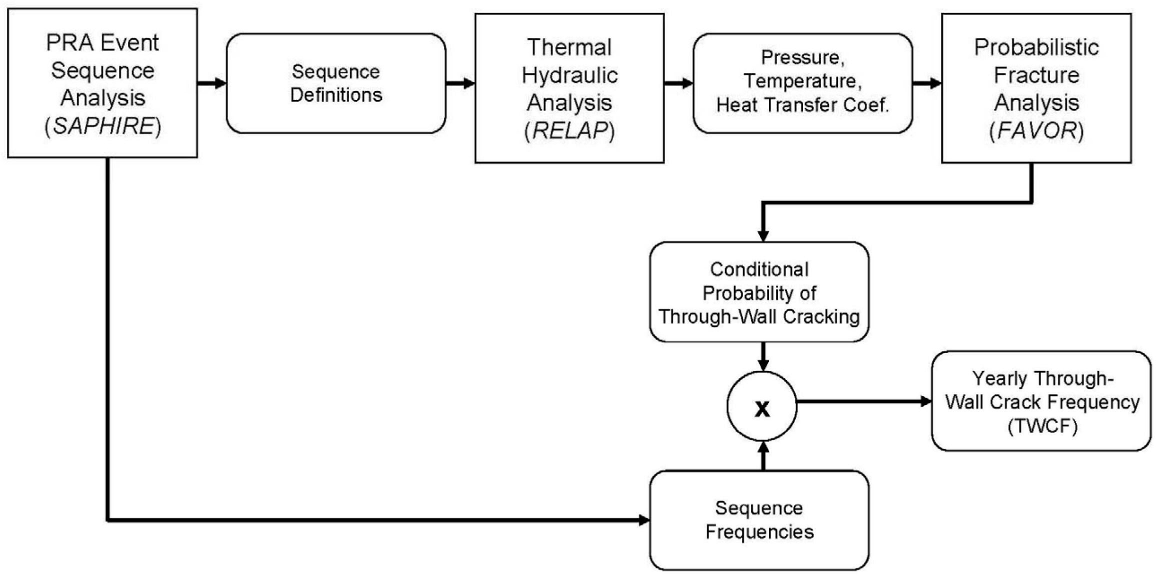

Fig. 3. PTS PRA Approach (adapted from [13])

pressurization and rapid RPV cooling, stress the RPV and potentially cause a PTS-induced brittle fracture.[12] For representative event sequences, thermal hydraulic (T/H) analyses were performed to estimate the temporal variation of temperature, pressure, and heat-transfer coefficient in the RPV downcomer. These temperature, pressure, and heat-transfer coefficient histories were then passed to a probabilistic fracture mechanics (PFM) model to estimate the conditional probability (for each sequence and for specified points of time in the plant lifetime) that a pre-existing flaw in the RPV could grow large enough to penetrate through the RPV wall. The result of the analysis was an estimated annual frequency of through-wall cracking at different points of time.

This approach is similar to that used in a set of PTS PRA studies that were completed in the mid-1980s [98-101]. However, the new PTS PRAs, in addition to using the latest available data on initiating event frequencies and equipment failure probabilities: (1) explicitly analyzed a larger number of event sequences, (2) used a modern approach to address operator actions, and (3) provided a full, quantitative treatment of uncertainties.

Regarding the number of event sequences, the older studies identified a large number ($\sim 10^5$) of potential PTS sequences, but gathered them into a small number of coarse groups (~10) due to then-current limitations in the ability to perform multiple T/H calculations. The coarse groups were characterized conservatively, using the most challenging event sequence in the group to represent the effect of all sequences in the group. The new PTS PRA studies also grouped sequences, but did this in an iterative manner. Groups of sequences found to contribute significantly to TWCF were subdivided into smaller groups until the TWCF estimate stabilized. As a result of this process, the new studies explicitly analyzed about 50-100 representative sequences, and therefore provided a more finely detailed representation of the sources of PTS risk. Moreover, the analysis process ensured that the grouping process was not significantly affecting the study's numerical results.

Regarding the treatment of operator actions and HRA, the old PTS PRA studies used either THERP or STAHR (Sociotechnical Assessment of Human Reliability), an HRA method that uses: a) influence diagrams to represent the effect of PSFs on HEPs, and b) facilitated sessions to quantify the strength of these effects. The new PTS PRA studies used the ATHEANA method. As discussed in Section 3.2.2, in contrast with older HRA methods, ATHEANA emphasizes the identification of error forcing contexts, and provides a search process to do this (see Step 6 in Figure 2). Without information such as that being developed by the HRA Empirical Study [87], it cannot yet be demonstrated which HRA approach is a better predictor of human error. However, the ATHEANA approach is more consistent with current views on how human error should be addressed in a PRA.

Regarding the treatment of uncertainties, the old PTS PRA studies used point estimates and dealt with uncertainties using conservative assumptions and sensitivity studies. In contrast, the new PTS PRA study

[12] Note that unlike most nuclear power plant PRAs, a PTS PRA is concerned with events that initially involve overcooling of the core. Fuel overheating and damage occur only after the RPV cracks and primary coolant escapes the reactor coolant system.

generally attempted to use realistic input values and models and quantified the uncertainties in the model results. The study also distinguished between aleatory and epistemic uncertainties in all phases of the analysis. This distinction proved useful to the analysis team when integrating the T/H and PFM phenomenological analyses into the PTS PRAs.

As discussed in NUREG-1806, the results of the PTS Reevaluation Project indicate that the risk from PTS events is extremely low (and much lower than previously calculated). For the plants studied, the mean TWCF is estimated to be on the order of 10^{-8}/yr or less after a postulated 60 years of operation (i.e., operation under a potential plant license extension).[13] NUREG-1806 also provides a qualitative analysis that indicates that even should a PTS event induce a through-wall crack, the likelihood that this will lead to a large, early release of radioactivity is very small. Overall, the PTS Reevaluation Project confirmed that the technical basis underlying the current PTS Rule is very conservative.

3.3.3 Technical Accomplishments

Although the PTS PRAs performed for the PTS Reevaluation Project did not involve any new methodological advances, they did support the field testing and refinement of NRC's newer methods and approaches in the area of HRA and uncertainty analysis.

In the case of HRA, the project provided a venue for the first use of ATHEANA in PRAs developed to support regulatory decision making. ATHEANA was used to identify, model, and quantify all (~15-30, depending on the plant) of the HFEs in the PTS PRAs.[14] ATHEANA also helped the analysis team identify and treat a number of errors of commission [102].

In the case of uncertainty analysis, the PTS Reevaluation Project spent considerable effort to implement the aleatory/epistemic framework discussed by Apostolakis [103]. The project: (1) considered all potentially significant sources of uncertainty in the PRA event sequence models, the T/H models, and the PFM models; (2) categorized these sources as being aleatory or epistemic; (3) treated the aleatory uncertainties in the models for TWCF; and (4) propagated some of the epistemic uncertainties through the models for TWCF. Due to practical concerns, some of the epistemic uncertainties in the T/H and PFM analyses were addressed using sensitivity studies or conservative modeling approaches. The implementation approach discussed in NUREG-1806, including the allowances for practical concerns, may prove to be useful for other risk-informed treatments of engineering issues.

3.3.4 Regulatory Impact

Based on the results documented in NUREG-1806, NRC initiated a rulemaking process to provide a voluntary alternative rule that licensees may elect to implement instead of the existing PTS Rule. The existing rule will remain in the Code of Federal Regulations. A draft version of the new rule was issued for public comment in 2007. It is currently expected that the final version of the new rule will be published in 2009.

3.4 Summary Remarks - Case Studies

In the preceding sections, we have reviewed three cases in which PRA research at the NRC has supported the implementation of the agency's 1995 PRA Policy Statement [9]. In the case involving fire PRA, research was triggered by a major event (the Browns Ferry fire). In the case involving HRA, research was performed to support the original WASH-1400 study and then to address review comments on that study. The TMI accident highlighted the need for improved HRA methods and indicated specific issues requiring treatment. In the case involving PTS, research was triggered by non-PRA research results that indicated a potentially fruitful area for regulatory action.

Despite the differing trigger events, issues of interest, and underlying phenomena, these cases have a number of common technical features. In each case, work was needed to go beyond a simple, statistically-oriented treatment of PRA basic events. This work involved the development of an improved understanding of key phenomenology, the creation of a mathematical framework to incorporate this understanding in a PRA, the development of models and tools to implement this framework, and the testing of the models and tools in practical decision support applications. The importance of non-operational data is another common factor in each case. Data from fire experiments have been used to validate the fire models used in fire PRAs and to support assessments of the likelihood of particular fire-induced cable failures. Data from separate-effects and integral tests have been used to assess the T/H models used in the PTS studies. Data from materials tests have been used to develop and calibrate the PFM models used in the PTS studies. Data from the Halden Reactor Project's control room simulator facility are being used to benchmark HRA models.

From a regulatory applications viewpoint, although all three cases have had a positive impact on NRC's regulatory activities, it is worth noting that in two of the cases, the research was originally aimed at improving the

[13] The uncertainty distributions for TWCF have a large spread, typically spanning two to three orders of magnitude. The mean TWCF value generally corresponds to the 90^{th} percentile (or higher) of the distribution.

[14] Note that ATHEANA was being refined during the PTS Reevaluation Project. Thus, there were some differences in the HRAs performed for the three PTS PRAs, especially regarding the quantification of HEPs. These differences were not large enough to affect the overall conclusions of the PTS Reevaluation Project.

agency's state of knowledge regarding key sources of risk rather than changing NRC's regulatory approach to specific issues. Furthermore, in all three cases, the research required multi-year efforts, and the effects of research were felt years (even decades) after the research was initiated.

For example, regarding fire PRA, research started after Browns Ferry fire took about three years to develop methods and tools suitable for use in an actual PRA. It took nearly another 25 years of development and application before fire risk considerations were formally integrated (through standards and rulemaking) into the agency's fire protection requirements. In the case of PTS, where the PTS Reevaluation Project was specifically aimed at developing the technical basis for a potential rule change, the main body of research was completed and documented roughly 8 years after the start of work, and it is expected that the final alternative rule will be published two years after that.

4. CONCLUDING REMARKS

NRC's PRA research plays an essential role in support of the agency's implementation of its 1995 PRA Policy Statement. This research provides the technical basis for NRC's regulatory activities in key areas; providing PRA methods, tools, and data enabling the agency to meet future challenges; supporting the implementation of NRC's PRA Policy Statement by enabling the assessment of key sources of risk; and supporting the development of technical and human resources needed to support NRC's risk-informed activities.

Through the use of three case studies, this paper demonstrates that:

- PRA research initially aimed at improving the agency's understanding of an issue can positively (and significantly) affect NRC's regulatory processes;
- Successful PRA research efforts can take years to complete; and
- The regulatory impact of PRA research may not be felt for many years.

The first two points indicate the potential value of a PRA research program that, in addition to addressing current regulatory needs, also includes sustained, longer-term activities aimed at improving the agency's state of knowledge regarding key sources of risk for current and potential future facilities. The last two points indicate that substantial lead time may be needed to fully address issues for which a technical basis does not yet exist.

Additionally, based on a consideration of the general objectives of NRC's regulatory research, this paper identifies a number of unexplored PRA topics and themes where research may be valuable. Currently, the NRC is developing a PRA research plan that considers such activities in the context of future challenges and agency needs [104].

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