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United States Nuclear Regulatory Commission

Protecting People and the Environment

A Short History of Fire Safety Research Sponsored by the U.S. Nuclear Regulatory Commission, 1975-2008

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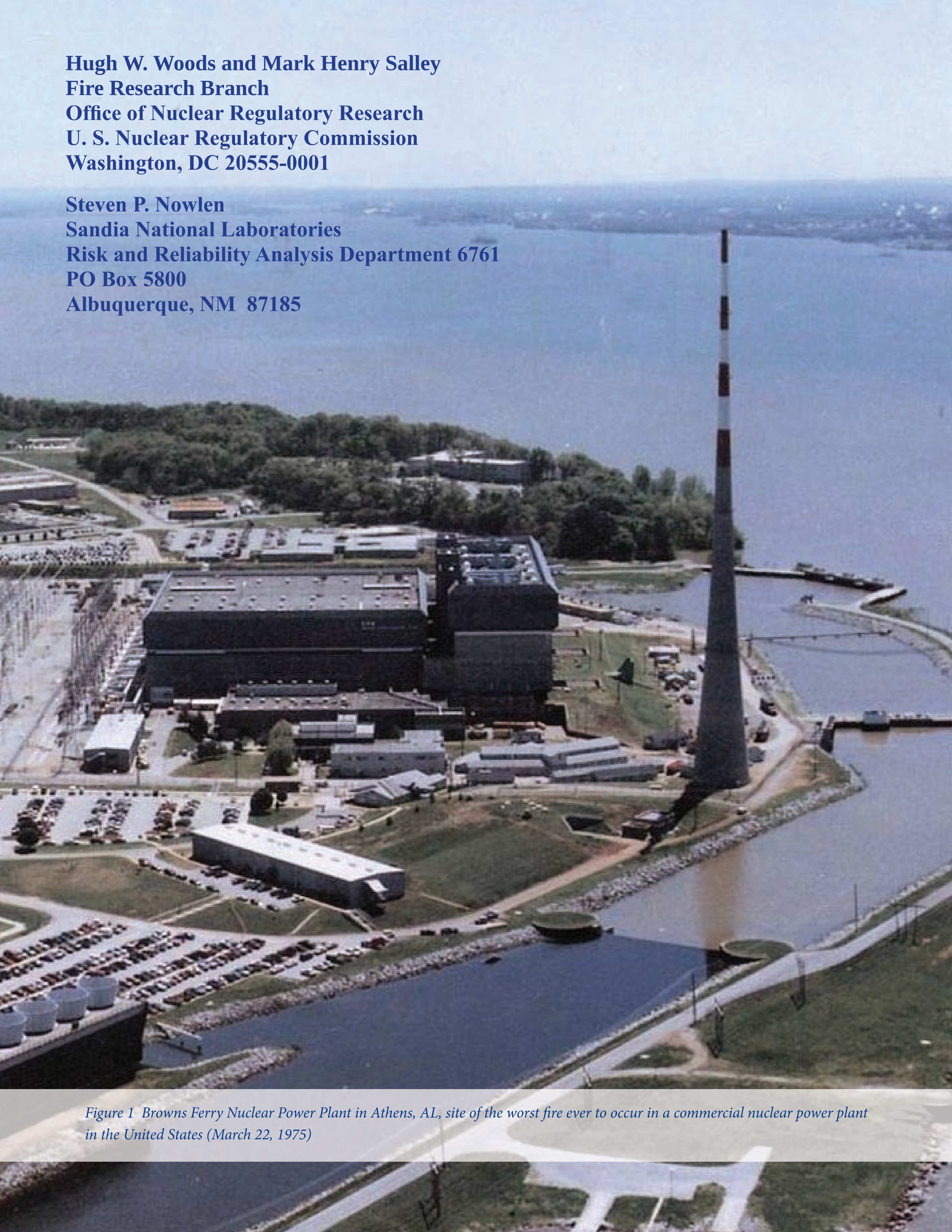

Figure 1 Browns Ferry Nuclear Power Plant in Athens, AL, site of the worst fire ever to occur in a commercial nuclear power plant in the United States (March 22, 1975)

Preface

This report summarizes research conducted by the U.S. Nuclear Regulatory Commission (NRC) from its inception through 2008, to identify and reduce challenges to the ability of a nuclear power plant (NPP) to safely shut down when necessary to prevent damage to its nuclear core due to indirect effects of the fire, such as fire-damage to its emergency core cooling systems.

The NRC was created on January 19, 1975, by the Energy Reorganization Act, which abolished the Atomic Energy Commission (AEC) and replaced it with the NRC, the Energy Research and Development Administration, and the Energy Resources Council. The latter two agencies later became part of the U.S. Department of Energy (DOE), which was created on October 1, 1977.

Thus the AEC, which had been charged with regulation and promotion of the nuclear power industry, was replaced by the NRC (which inherited only the regulatory function) and DOE (which inherited the promotional function).

When the NRC was created, regulation of the nuclear power industry was based primarily on a set of deterministic rules. Those rules, including the ones related to fire safety, relied heavily on the design requirements for large, nonnuclear industrial facilities. They did not rely on quantitative safety evaluations, such as probabilistic risk assessments (PRAs). The NRC did not publish the first PRA, WASH-1400, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," (NUREG-75/014) until October 1975.

However, three events happened early in the NRC's existence that accelerated a trend toward the increased use of quantitative methods in NPP regulation. The first was a major fire at the Browns Ferry Nuclear Power Plant (Figure 1, inside front cover) on March 22, 1975. This accident focused attention on fire as a "common mode" cause of multiple safety system failures, a factor that had not been adequately considered by the deterministic criteria used in the plant's design. The second event was the October 1975 publication of WASH-1400, which showed that quantitative safety evaluations were feasible and could be used to improve NPP safety. The third event was a severe accident at the Three Mile Island Unit 2 NPP on March 28, 1979.

In 1995, the NRC formalized this trend toward the increased use of quantitative methods in NPP regulation in the following Commissioners' policy statement:

> The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

This report covers the most significant NRC-sponsored fire-safety research programs over a time period from the agency's creation, when regulation was based primarily on deterministic (nonquantitative) rules, to the present time, when regulation is rapidly becoming "risk-informed," using a combination of quantitative and deterministic considerations.

This report is part of the Knowledge Management Program being conducted by the NRC's Office of Nuclear Regulatory Research, Division of Risk Analysis, Fire Research Branch.

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List of Acronyms

AEC	Atomic Energy Commission
AEOD	NRC's Office for the Analysis and Evaluation of Operational Data
AHJ	Authority Having Jurisdiction
BFN	Browns Ferry Nuclear Power Plant
BNL	Brookhaven National Laboratory
CAROLFIRE	Cable Response to Live Fire
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DC	Direct Current
DESIREE-FIRE	Direct Current Electrical Shorting In Response to Exposure-Fire
DOE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
EIS	Electrical Isolation Scheme
EPRI	Electric Power Research Institute
ERFBS	Electrical Raceway Fire Barrier Systems
FD&S	Fire Detection and Suppression
FEDB	Fire Event Database
FIVE	Fire-Induced Vulnerability Evaluation
FPRP	Fire Protection Research Program
FPS	Fire Protection System
FRSS	Fire Risk Scoping Study
GI	Generic Issue
GL	Generic Letter
HEP	Human Error Probability
HFE	Human Failure Event
HLF	Heat Loss Factor
HRA	Human Reliability Analysis
HRR	Heat Release Rate
IEEE	Institute of Electrical and Electronics Engineers

IN	Information Notice
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Examination of External Events
LLE	Liquid-Liquid Extraction
LOOP	Loss of Offsite Power
MCR	Main Control Room
MFFF	Mixed Oxide Fuel Fabrication Facility
MOU	Memorandum of Understanding
MOX	Mixed Oxide (i.e., Mixed Uranium Oxide and Plutonium Oxide)
MT	A particular 3 hour fire barrier material's name (not an acronym)
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards (NRC)
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
RES	Office of Nuclear Regulatory Research (NRC)
RG	Regulatory Guide
RI/PB	Risk-Informed, Performance-Based
RMIEP	Risk Methods Integration and Evaluation Program
RSP	Remote Shutdown Panel
SISBO	Self-Induced Station Blackout
SNL	Sandia National Laboratories
THIEF	Thermally-Induced Electrical Failure
V&V	Verification and Validation

Chapter 1: Background, Organization, and Scope

This report summarizes research conducted or sponsored by the U.S. Nuclear Regulatory Commission (NRC) from its inception through 2008, to identify and reduce challenges to the ability of a nuclear power plant (NPP) to safely shut down when necessary to prevent damage to its nuclear core due to indirect effects of the fire, such as fire-damage to its emergency core cooling systems.

Depending on the design and operational characteristics of a given NPP, fire can be a significant or even dominant contributor to the overall probability of core damage at that plant. This assertion is substantiated by several core damage "near misses" caused by fires that have occurred in the United States and abroad (e.g., Browns Ferry Nuclear Power Plant (BFN), 1975; Vandellos, 1989; Narora, 1993). It is also supported by fire probabilistic risk assessments (PRAs) and individual plant examinations of external events (IPEEEs), many of which have estimated mean fire-induced core damage frequencies (CDFs) of 10-4/year or greater, predicted contributions to total CDF of greater than 20 percent, or both.

Because of the significant risk associated with fires at NPPs, the NRC has sponsored a considerable amount of research on fire-related issues. Although the nature and exact goals of this research have evolved over the years (as mentioned in this report's Preface), it has all been directed to support regulatory activities that ensure NPP fire safety and, in more recent years, that enable a quantitative assessment of the level of fire safety of an NPP.

This report summarizes the results of NRC-sponsored fire-safety research from 1975 (when the NRC was created) through 2008, a time period during which the NRC was in the process of transforming regulation from a system based primarily on deterministic (nonquantitative) rules to the present system which is more "risk- informed," based on a combination of quantitative and deterministic considerations (this transformation process is continuing beyond 2008). This report provides a historical overview of the major research projects during that period of transformation, in the following four phases:

(1) the Fire Protection Research Program (FPRP), 1975–1987

(2) from termination of the FPRP until completion of the LaSalle Risk Methods Integration and Evaluation Program (RMIEP), 1987–1993.

(3) from RMIEP completion to the start of recent projects (1993–1998)

(4) recent projects (1998 through 2008)

In discussing NRC-sponsored fire-safety research, this report covers a wide variety of activities performed by its Office of Nuclear Regulatory Research (RES), the Office of Nuclear Reactor Regulation (NRR), and the former Office for the Analysis and Evaluation of Operational Data (AEOD). This coverage includes experimental investigations of fire and the behavior of components exposed to fires, surveys of plant practices, assessments of the safety implications of these practices, the development of analytical methodologies to perform such assessments, and a review of fire PRAs performed by other organizations (e.g., IPEEE submittals from NPPs). It also discusses the potential influence of other safety issues on future PRA improvements (e.g., eight fire-related generic issues [GIs] are included). And, although the report's subject is NRC-sponsored fire-safety research, where industry- or university-sponsored activities are an essential part of that research, those activities are also included (e.g., industry-sponsored PRAs that were the first to use the NRC-sponsored COMPBRN fire model).

By discussing these varied topics, this history report provides an overview of the many sources of fire risk from NPP operations, as well as what has been done in the past (and is planned for the future) to lessen that risk.

Chapter 2: Historical Overview—the Four Phases of NRC Fire Research

This report presents the NRC research program regarding fire safety in NPPs in terms of the following four phases, which Chapters 3 through 6 discuss in more detail.

The first phase (1975–1987) began with the initiation of the FPRP. The BFN fire occurred early in that period, on March 22, 1975, causing the development and imposition of a whole new set of fire protection requirements.[1] A principal goal of the FPRP in its early years (1975–1983) was thus confirmation of the effectiveness of the new requirements (such as those related to the operability of safety-related equipment during a fire). In 1983, the NRC significantly redirected the FPRP, changing its goals to developing the test data and analytical capabilities needed to determine NPP fire risk, determining fire effects on control room equipment and operations, and determining the effect of suppression system actuation on safety equipment.

The second phase (1987–1993) covered post-FPRP activities designed to assess the importance of a set of topical issues that had not been included in previously performed fire PRAs; the technical basis for resolving GI 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"; and the fire-risk analyses of several NPPs. This phase ended with completion of the LaSalle fire PRA, part of the RMIEP, which, in 1993, applied the latest quantitative risk analysis technology in a large-scale study.

The third phase (1994–1998) covered developments after completion of the RMIEP regarding the effect of smoke on digital components, the performance of penetration seals, turbine building fire hazards, then-current operational experience data with respect to fires, the risk implications of safe-shutdown methods (in the event of a serious fire) that cause station blackout,[2] the resolution of eight GIs as discussed in this document, and a review of the IPEEE submittals from NPPs.

The fourth phase started in 1998, after the Commission's 1995 policy statement encouraged the use of PRA methods wherever possible to support NPP regulation. The activities in this phase each support one or more of the four functional areas of fire PRAs—fire prevention, fire detection and suppression, fire mitigation, and the quantitative evaluation of fire safety.

Before presenting a detailed historical overview of the early phases of the NRC's fire research program, especially the FPRP, it is necessary to discuss the controlling influence the BFN fire had on fire-safety regulations. The BFN fire was of such significance that it resulted in a new generation of rules and regulations regarding fires at NPPs. In its early years, the FPRP provided the research required to confirm certain underlying assumptions on which the NRC based its new fire rules and regulations. Thus, this report discusses the BFN fire, along with the requirements resulting from that fire and the regulatory research carried out to confirm the most significant assumptions underlying those requirements.

The Browns Ferry Fire and Appendix R

When the U.S. nuclear industry designed (and built, in many cases) the current generation of

[1] Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to Title 10, of the Code of Federal Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities (10 CFR Part 50).

[2] "Station Blackout" is the loss of offsite electric power plus loss of backup power from the diesel generators.

commercial NPPs, the NRC stated its fire protection requirements in terms of broad performance objectives for the design and location of systems, structures, and components important to safety; the use of noncombustible and heat-resistant materials; and the provision of fire detection and suppression systems. No detailed implementation guidance existed to determine whether a plant's fire protection program met these objectives, and the NRC staff relied upon compliance with local fire codes and insurance underwriter ratings to determine acceptability. These codes and ratings primarily addressed life safety and property loss prevention, and the nuclear industry initially used them because few plant designers recognized that fires could represent a significant threat to safe plant operations.

However, it is now recognized that a fire in an NPP can induce the failure of multiple plant safety and support systems and thereby represents a potential threat to the integrity of the reactor core. The severe cable tray fire at BFN on March 22, 1975, brought this concern to the forefront.

The BFN fire began when a worker used a candle to inspect for air leakage during the installation of temporary penetration seals on fire-barrier cable trays. A ventilation-induced differential pressure between the plant cable spreading room and the Unit 1 reactor building was being used as the driving force for smoke from the candle. The movement of smoke toward and through the seal would indicate a crack in a temporary penetration seal. During this inspection, the candle flame set the temporary polyurethane penetration seal material on fire when it was sucked into the crack. The fire quickly spread to the cables on both sides of the penetration and burned uncontrollably for almost 8 hours.[3] During the fire, the control room received numerous erroneous instrument readings and spurious indications of systems starting and stopping (of course, at the time, the operators did not know if they were real or not). In large part, the continued burning resulted from reluctance on the part of plant operators to apply water to the fire for fear of shorting out vital electrical safety systems. Once water was applied, the fire was quickly brought under control. The fire damaged over 1,600 electric cables and rendered all of the Unit 1 and many of the Unit 2 emergency core cooling systems (ECCS) inoperable. Figure 2 shows BFN cable trays with cables damaged by the fire.

Although the BFN fire disabled a significant number of plant safety systems, the operators successfully brought the reactor from power operation to a safe

Figure 2 Browns Ferry cable trays after the fire, showing damaged cables

[3] Previous fires had been started in this way on several occasions, but each time, the workers had been able to beat out the flames with a flashlight. Unfortunately, those clear warnings of the danger were ignored.

shutdown condition. However, the loss of multiple safety systems resulted in significant difficulties in achieving a safe shutdown state. Operators had to initiate a number of untested recovery actions to restore plant systems and achieve a stable reactor condition.

Fire-Related Requirements Resulting from the Browns Ferry Fire

As a direct result of the near-miss accident at BFN, the NRC formulated a new set of fire-safety requirements to be applied to all commercial U.S. NPPs. The NRC developed these requirements, formalized as Appendix R, to ensure the continued availability of a plant's safety-related functions during and after a fire. They specifically addressed the issues of plant safety system operability in a fire situation (i.e., those issues that had not been addressed by general industry standards). These fire safety requirements went into effect in November 1980, with three sections (III G., J., and O.) that applied retroactively to all units licensed to operate before January 1, 1979.

In general, Appendix R requirements specify alternative methods for protecting (through separation and other means) redundant trains of plant safety equipment. Because these requirements were applied as a retrofit, they specifically allowed for the housing of redundant safety systems within a single fire area. A fire area is a region bounded on all sides by fire barriers with ratings that are commensurate with the hazard; fire zones are a subset of fire areas and can be delineated by lesser barriers. The most controversial—and least restrictive—of the alternative redundant train separation criteria identified in Appendix R are the so-called "twenty-foot (6.1 meter) separation criteria" by which:

- 20 feet (6.1 meters) of horizontal space must separate redundant equipment, with no intervening combustibles, and
- automatic fire detection and suppression systems must protect the area.

Appendix R requirements also specify that actions be taken regarding other aspects of plant design and operation. These additional requirements include installation of a remote shutdown station, which is physically and electrically independent of the main control room (MCR); the creation and training of manual fire response teams; the use of low flame-spread cables qualified by the Institute of Electrical and Electronics Engineers (IEEE) 383 1974 Flame Spread Test for all new installations; and the creation of a fire protection management structure to formalize fire protection practices.

NUREG/BR-0361 and its enclosed DVD contain all the information that could be assembled about the fire and its effects on NPP regulations.

Chapter 3: The Fire Protection Research Program and Other Fire-Safety-Related Activities, 1975–1987

The NRC recognized that the effectiveness of Appendix R requirements in ensuring the availability of the plant's safety-related functions during a fire depended on the correctness of certain underlying assumptions regarding redundant equipment separation (e.g., electric cables); automatic suppression systems, including water and other fire-extinguishing agents; automatic fire detection systems; fire shields, barriers, and cable coatings; the flammability of older electric cable insulation; fire effects on safety-related equipment other than cables; and the use of low flame-spread cables in new installations. Confirming the correctness of these assumptions became the original goal of the 1975–1987 FPRP.

Sandia National Laboratories (SNL) has been active in fire-related research for several decades, particularly in the areas involving fire protection and fire risk in NPPs, such as (but not limited to) the FPRP. Because SNL participated in many of the fire research activities discussed in this fire history, its involvement with each may not be specifically mentioned, although the contributions of other organizations and individuals are often cited where appropriate.

To support its changing regulatory needs, the NRC expanded the goals of the FPRP in Fiscal Year (FY) 1983 to include developing test data and the analytical capabilities needed to determine NPP fire risk, determining fire effects on MCR equipment and operations, and determining the effects of suppression system actuation on safety equipment. The next-to-last section of this chapter, "Expansion of the FPRP's Goal," discusses these items.

In 1977, in parallel with the FPRP, the NRC initiated a research project at the University of California at Los Angeles (UCLA) to develop a method for estimating fire risk at NPPs. The UCLA project was later supplemented by an NRC-sponsored project at the Rensselaer Polytechnic Institute (RPI). These projects resulted in the development of a method and associated tools (including the fire-modeling computer code COMPBRN) that were used to conduct a number of industry-sponsored PRAs, including those at Zion (1981) and Indian Point, (1982), each of which explicitly addressed fire risk. Also, Brookhaven National Laboratory (BNL) adapted an additional fire model for use on NPP fires and demonstrated preliminary applications. The last section of this chapter, "Parallel Programs Not Part of the FPRP," discusses the UCLA, RPI, and BNL efforts.

The Effectiveness of Redundant Equipment Separation

One of the principal requirements of Appendix R is that, in an area containing redundant trains of equipment, licensees must maintain a separation between the trains of 20 feet (6.1 meters) horizontally with no intervening combustibles, and the area must also include an automatic fire detection and suppression system. Therefore, an investigation of the adequacy of the 20 foot (6.1 meter) separation criterion was initiated. Under contract with SNL, Underwriters Laboratories, Inc., Northbrook, IL, conducted preliminary experiments and full-scale tests. The tests demonstrated that, for a small room where the effects of hot gas layers could become significant, a separation of 20 feet (6.1 meters) was not, in and of itself, sufficient to ensure that cabling so separated from the source fire would remain undamaged. However, these tests did not take into account the additional measure of safety afforded in such situations by the requirement for an automatic fire detection and suppression system.

Additional tests placed nonpressurized sprinkler heads within the test enclosure and monitored for activation of their fusible link. These tests, which involved simulated fire suppression, found that activation (near the test-measured cable damage time) of a suppression system that performed as designed would prevent the observed cable damage.

Automatic Fire Suppression Systems

An experimental program assessed the performance of carbon dioxide, Halon, and water-based sprinkler systems. These tests demonstrated that all the suppressants could effectively contain fully developed cable-tray fires, when installed according to existing general industry practices.

Tests found that directed water-spray suppression was the most effective system for extinguishing and preventing reignition of all fire sizes, cable types, and tray configurations tested.

For gaseous systems, tests showed that prolonged soak times (15 to 20 minutes) at full concentration were necessary to ensure that deep-seated cable fires would be extinguished (it was noted that industry standards at that time included the consideration of soak times for deep-seated fires).

Automatic Fire-Detection Systems

Several analytical studies assessed the performance of fire-detection systems and provided guidance on the design, installation, and maintenance of such systems in an NPP application. These studies concluded that general industry standards for fire-detection systems were broadly applicable to NPP applications but that certain NPP applications involved more congested conditions than those encountered in the general industry. Consequently, the studies recommended in-situ testing of NPP detection systems. Although preliminary methodologies for the design and installation of fire-detection systems were developed, confirmatory testing of detector performance resulting from the application of those methods was not undertaken because studies concluded that in-situ testing of detectors would be needed in each plant area.

Effectiveness of Fire Shields, Fire Barriers, and Fire-Retardant Cable Coatings

The results obtained from the above activities led to research to test the effectiveness of additional fire-protection measures for cable trays, such as fire shields, fire-retardant cable coatings, and fire barriers.

Experimental studies at SNL investigated the effectiveness of various cable tray shields, including ceramic blankets wrapped around a cable tray, solid tray covers and bottom plates, ventilated tray bottom plates, and noncombustible mineral fire boards between trays. The studies found that all the shield systems provided some measure of added protection, although performance varied widely and no system ensured the prevention of fire propagation or fire-induced damage.

SNL studies also investigated the effectiveness of various retardant coating products. Although some delays in the propagation of fires, fire damage, or both, occurred for some of the tested products, the coating performances varied widely (one product actually increased the fire's severity). No coatings were adequate to ensure that fires would not propagate nor could they prevent fire damage under exposure fire conditions.

The SNL examination of fire barriers studied and evaluated standards for the qualification of fire-barrier elements and performed thermal analyses of typical 3 hour fire-barrier systems to determine their response under various conditions.

The study concluded that guidelines for the qualification of fire-barrier elements provided for adequate exposure fire intensity and recommended no changes in that area. However, it identified two areas of potential weakness. The first was that the hose stream part of the test had poor repeatability and was imprecise and difficult to control. The second was the failure of U.S. standards to incorporate a positive furnace pressure requirement during fire-exposure testing. This shortcoming in the test procedures could result in nonconservative estimates of fire-barrier endurance under actual exposure conditions. For example, a negative furnace pressure during a test would tend to draw cooler air from outside the furnace into the volume behind the barrier (i.e., onto the protected cables). But in an actual fire, a positive pressure in the fire area might instead force hot air through the barrier onto the protected cables. Thus, the results from a negative furnace pressure test might not conservatively bound the higher temperatures on the protected cable that might result from an actual fire event. The resolution of GI 149, "Adequacy of Fire Barriers," noted this failure of U.S. fire-barrier testing standards to require a positive furnace pressure and stated that, among the industrialized countries, only the United States continues to endorse neutral or negative pressure fire-barrier testing.

Flammability of Older Electric Cable Insulation

Most fire tests involving electric cables have been conducted using new cables purchased for the tests. However, because electric cables in NPPs are aged to varying degrees, depending on the age of the plant, the aging of cable insulation and jacketing materials might result in an increased fire hazard. Preliminary studies conducted as part of the FPRP found that, because fire retardant additives used in cable manufacture were not lost during the aging process, and because polymer aging is largely an oxidation process, aging would likely decrease material flammability. Later tests confirmed that cable aging significantly reduced flammability and that, for purposes of fire-safety analysis, the use of flammability parameters based on new cables would be appropriate for the full 40 year anticipated life of an NPP.

Fire Effects on Safety-Related Equipment Other Than Cables

Fire damage to electric cables is a major contributor to the overall fire-related CDF in NPPs. Electric cable insulation, which is distributed throughout the plant, is one of the largest masses of flammable material in an NPP. Consequently, its damage by fire can cause associated safety equipment to become inoperable or to spuriously operate (which could damage additional plant systems). In addition, a fire environment can damage the safety equipment itself. The term, "fire environment," includes the effects of heated air, radiant heat flux, moisture, smoke, and corrosive species (e.g., combustion products dissolved in water used to combat the fire).

A review of vendor information, fire damage reports, equipment qualification tests (including fire test results), and material properties, identified 33 types of components found in NPPs and ranked them in terms of their potential sensitivity to fire environments, considering both their functional requirements and propensity for damage. Based on this review, relays and hand switches were selected as first choices for fire damage testing, and logic equipment, power supplies, transmitters, and motor control centers were identified as future candidates for testing.

Most of the fire tests positioned the components so that they were not involved in the actual fire. Thus, the environments were described as secondary fire environments and did not include such effects as flame impingement or actual component burning. These experiments provided numerous detailed insights for use in future NPP fire-risk analyses.[4]

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[4] SAND90-1827, "Fire Safety Lessons Learned from the Design and Operation of Commercial Nuclear Reactor Facilities," February 1993, pp. 57, 58, 61, and 62; and NUREG/CR-5384, "A Summary of NPP Fire Safety Research at SNL," December 1989, pp. 119–121.

Effectiveness of Using Low Flame-Spread ("Qualified") Electric Cables

Appendix R requirements specify the use of low flame-spread cables for all new installations (i.e., cables "qualified" by passing the IEEE 383 1974 Flame Spread Test must be used). Although refitting of existing cables is not required, all new installations and cables replaced as a part of maintenance activities must be qualified.

Fire tests conducted as part of the FPRP found that qualified electric cables are more difficult to ignite and spread fire more slowly than nonrated cables; however, they can be ignited, burned, or damaged. The fire tests also observed that, once a self-sustaining fire is established in qualified cables, it tends to be more intense and more difficult to extinguish.

Expansion of the FPRP's Goals

In 1983, the NRC significantly expanded the objectives of the FPRP to include developing test data and the analytical capabilities needed to determine NPP fire risk, determining fire effects on control room equipment and operations, and determining the effects of suppression system actuation on safety equipment. This expansion represented an important step in the NRC's increased use of quantitative methods in NPP regulation.

The following threefold research approach was planned to assist in reaching these expanded goals:

(1) Define Fire Sources. Characterize a range of fire sources with respect to energy and mass evolution, including smoke, corrosion products, and electrically conductive products of combustion.

(2) Define Environments. Develop an analytical method for determining the environment resulting from fire; account for the source characteristics, the suppression action following detection of the fire, and certain parameters specific to the plant enclosure in which the fire originates, such as the geometry of the enclosure and the ventilation rate; and describe the developing local environment in the vicinity of safety-related equipment in terms of temperatures, temperature rise rates, heat fluxes, and moisture and certain species content.

(3) Define Equipment Response. Study the response of certain safe-shutdown equipment and components to the environmental conditions to determine the limits of environmental conditions to which a component may be exposed without impairing its ability to function.

Arguably the most notable part of the above threefold program was the 25 large-scale fire tests (the "Fire Model Validation Tests") conducted by SNL. These tests collected data from fires with a range of fuels, fire intensities, fire locations, and ventilation conditions. The data supported fire model improvements (as documented in NUREG/CR-4681, "Enclosure Environment Characterization Testing for the Base Line Validation of Computer Fire Simulation Codes," issued March 1987; NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets," Part I: "Cabinet Effects Tests," Volume 1, issued April 1987; and Part II: "Room Effects Tests," Volume 2, issued November 1988).

Development and evaluation of the above threefold approach was coordinated with a parallel program then in progress on control room habitability and with the RMIEP. The RMIEP was a PRA method then being planned and conducted to evaluate "all risks" (including those resulting from fire) associated with the La Salle NPP and to demonstrate the use of the improved PRA methodology available at that time.

Although the NRC terminated the FPRP in FY 1987, before some of its tasks were completed, it included the results from those tasks that had been completed in planning the RMIEP, which took place during the 1987–1993 time period covered in Chapter 4.

Parallel Programs Not Part of the FPRP

In parallel with the FPRP, the NRC sponsored a project to develop a fire PRA methodology at UCLA, with supplemental efforts at RPI and BNL. The resulting methodology was used to conduct a number of industry-sponsored fire PRAs, including those at Zion and Indian Point, which then received intensive NRC-sponsored reviews at BNL and SNL. Those industry-sponsored fire PRAs were thus the first to apply the UCLA-developed framework, which was later used in NRC-sponsored fire PRAs, including the RMIEP (LaSalle) and the plants in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants—Final Summary Report," issued December 1990, discussed in the next chapter. The industry also sponsored other fire PRAs during this time period.[5] The following sections present further details of the UCLA, RPI, and BNL efforts.

UCLA—In 1977, the NRC initiated a new research project, led by G. Apostolakis at UCLA, to develop a methodology for assessing fire risk at NPPs (the project was intended to complement the NRC's FPRP, discussed above, and was later supplemented by NRC-sponsored efforts at RPI and BNL, discussed below).

The Apostolakis team developed and demonstrated an approach that integrated the predictions of a deterministic computer model for fire behavior (i.e., COMPBRN, developed by N. Siu at UCLA) into the assessment. The team accomplished this integration through the use of a competing-risks framework that computed the probability of fire damage to equipment (including electrical cables) as the outcome of a "race" between two simultaneous processes, fire growth and fire suppression.

The major elements of this framework were the COMPBRN fire model to predict the environment in the area (and the time to damage for target(s) in that area), and a probabilistic model to develop the distribution of the times needed to suppress the fire. The framework explicitly addressed uncertainties in the predictions of the physical models and in the parameters of the suppression model.

A number of industry-sponsored PRAs, including those at Zion (1981) and Indian Point (1982), used the UCLA-developed approach, which showed that fire could be an important contributor to CDF and risk because of its potential to act as a failure mechanism affecting multiple trains of equipment.

Thus, the NRC sponsored the method's development (at UCLA), and also the review and evaluation (at SNL and BNL) of its industry-sponsored applications.

RPI—RPI developed a methodology for evaluating the probability for loss of NPP safety functions because of fire. It established a framework for investigating fire scenarios that modeled fire development through its stages of ignition, detection, propagation, and suppression. RPI applied the methodology to a generic, or representative, boiling-water reactor (i.e., the plant characteristics assumed did not represent any specific plant) and obtained conservative estimates of core-damage probabilities from postulated fires. The RPI study also discussed variations in the methodology for application to specific plants.

BNL—BNL completed a survey of enclosure fire models that employed three-dimensional, transient field model techniques. It determined that one model (the PHOENICS code) could analyze

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[5] Other industry-sponsored fire PRAs included Big Rock Point, Millstone 3, and Oconee.

the fire-induced environment in enclosures typical of NPP critical areas. BNL compared its results with experimental measurements from two test programs conducted by the FPRP at the Factory Mutual Research Corporation fire-test facility in Rhode Island, one sponsored by the Electric Power Research Institute (EPRI), and the other sponsored by the NRC and conducted by SNL (i.e., the "fire model validation tests" discussed in the preceding section, "Expansion of the FPRP's Goals"). Comparisons with both test programs were promising; the model demonstration phase was essentially complete and model enhancement was planned. However, because of the termination of FPRP in FY 1987, BNL never fully validated the model, nor was it released for public use.

Chapter 4: The Fire Research Program Conducted Between Completion of the FPRP and Completion of the Risk Methods Integration and Evaluation Program, 1987–1993

During the period 1987–1993, studies assessed the following: the importance of a set of topical issues raised as a result of the FPRP that had not been included in previous fire PRAs; the technical basis for resolving GI 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"; and the fire risk at three plants, using an improved method that determined fire risk as part of a broader analysis of nonfire- and fire-related risk (i.e., the LaSalle fire PRA performed as part of the RMIEP, and the Peach Bottom Unit 2 and Surry Unit 1 fire PRAs reported in NUREG-1150). These studies are discussed below.

Topical Issues Raised as a Result of the Fire Protection Research Program

The Fire Risk Scoping Study (FRSS) investigated NPP fire-risk issues raised as a result of the FPRP. The specific objectives of the FRSS were to review and requantify fire-risk scenarios from four existing fire PRAs[6] in light of updated databases made available as a result of the FPRP, to use the updated computer fire modeling capabilities, and to identify and quantify (where possible) potentially significant fire-risk issues that had not been previously addressed in a fire-risk context.

The FRSS considered the following six issues: (1) control system interactions; (2) seismic-fire interactions; (3) the effectiveness of manual fire fighting (including smoke control); (4) total environment equipment survival (including the effects of the spurious operation of fire suppression systems); (5) the adequacy of fire-barrier qualification methods; and (6) the adequacy of analytical tools for fire-risk analysis. For most of these issues, the quantification of risk impact was hampered by a lack of experimental data for the associated phenomena, by a lack of sufficient plant-specific data upon which to base risk estimates, or both. However, the study made order-of-magnitude estimates of their relative risk importance. It identified control systems interactions and the effectiveness of manual fire fighting as having the most significant potential impact of the six issues; both were perceived as GIs that could raise the calculated CDF at many U.S. NPPs by an order of magnitude. Chapter 5 covers the period in which most of the work was performed regarding these topics (the section on "Electric Circuit Faults (Hot Shorts) and Self-Induced Station Blackout (SISBO)" and the section on "GI 148—Smoke Control and Manual Fire Fighting Effectiveness").

The Resolution of Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"

A number of precursor events showed that safety-related equipment subjected to fire protection system (FPS) water spray could be rendered inoperable. These events included numerous spurious FPS actuations initiated by operator testing errors or by maintenance activities (e.g., welding), or by steam or high humidity in the vicinity of FPS detectors or control circuitry. The effects of such events ranged from reactor trips to fires in high-voltage electrical

[6] These were limited scope requantifications of previous fire PRAs for Limerick, Indian Point 2, Seabrook, and Oconee.

equipment, and included water contamination of diesel fuel for the emergency diesel generators.

The NRC evaluated several representative NPPs and reviewed operational experience involving actuations of FPSs. It concluded that significant risks might exist at individual plants, and that cost-effective corrective actions might be possible to alleviate those risks. However, it also concluded that no such corrective actions could be identified that would be needed at all plants and that the identification of risks and corrective actions at any specific plant would require specific consideration of that individual plant (i.e., there were no truly generic concerns or generic corrective actions related to this issue). Thus, it concluded that the issue should be addressed on an individual plant basis.

Since the Individual Plant Examination of External Events for Severe Accident Vulnerabilities Program addressed the identified dominant risk contributors (discussed in Chapter 5), the resolution of GI 57 was subsumed into the IPEEE program.

Fire-Risk Assessment at Three Plants

The original quantification of core-damage risk at an NPP (WASH-1400, October 1975) only included events initiated by plant system and component faults, where the cause of failure was internal to the failed items (such events are called "internal events"). It did not include events initiated by earthquakes, floods, and fires, where the cause of failure is external to the failed items (such events were called "external events").[7] However, because of the fire at BFN in March 1975, the first fire PRA was performed in 1975 as a supplement to WASH-1400.[8] Its results provided a quick estimate of the risk implications of that fire, indicating that the CDF associated with the BFN fire was about 20 percent of the CDF due to the nonfire-related events that were addressed in the main body of the WASH-1400 study. It also recommended the development of a more detailed fire PRA method using improved models and data.

Consequently, when internal events were later quantified for six additional NPPs, three quantifications also included events caused by fires. The quantifications that included fire risk were Peach Bottom Unit 2 and Surry Unit 1, under the NUREG-1150 program, and LaSalle Unit 2, under the RMIEP program, NUREG/CR-4832, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)," Volume 9: "Internal Fire Analysis," issued January 1990.

The basic framework of the fire-risk quantification method applied in NUREG-1150 and RMIEP studies represented a milestone that has since become the accepted framework for conducting a "state-of-the-art" NPP fire-risk analysis. The framework is based on the full internal events PRA, with its event trees and fault trees providing consistency with respect to the internal events analysis, including the full gamut of random, test, and maintenance-related unavailabilities. This same framework is used in the currently recommended (2008) method presented in NUREG/CR-6850 (EPRI 1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005 (Chapter 6, section on "Development of a Joint NRC/EPRI Fire Risk Methodology").

Using this approach, the full benefit of the internal events analysis is obtained by performing that analysis first, then making changes where necessary to account for the unique effects of a fire. A fire can significantly change failure and error frequencies by providing a common cause that is not present for the internal events (e.g., by damaging electric cables controlling several systems or stressing the operators, thereby making multiple errors more likely).

7 When the WASH-1400 analyses were performed, "external" meant external to the component boundary. Since that time, "external" has evolved to mean external to the plant boundary. Therefore, by 2008, most fires were referred to as "internal" events.

8 Appendix XI to WASH-1400.

Chapter 5: Fire Research Conducted After Completion of the RMIEP, 1993–1998

During this period, the NRC and the nuclear industry undertook projects to better understand issues identified through a number of paths, including inspection results, operational experience, fire PRA results, and fire research results. The results of those projects have benefited industry and NRC programs by improving the PRA method used for the NUREG-1150 and RMIEP PRAs and enhancing the data available to support its applications. Examples include: the effect of smoke on digital components, the performance of penetration seals, the hazards of turbine-building fires, the risk implications obtained from further analysis of then-current operational experience fire data, the risk implications of safe shutdown methods used in the event of a serious fire causing station blackout, and the improved methods and data resulting from the resolution of certain GIs and from results of the IPEEEs. These issues are discussed below.

The Effect of Smoke on Digital Components

Research results showed that, in addition to the immediate effects of airborne smoke, a possible latent effect exists from the buildup of soot-like deposits on digital microelectronic circuits that could cause circuit failures even well after the fire is extinguished, including during the venting and purging period. Additional recommended work included tests to investigate the smoke effects on higher voltage alternating current (ac) equipment, such as switchgears, because the tests predominantly involved low-voltage direct current (dc) digital circuits.

Performance of Penetration Seals

Fire-barrier penetration seals are an element of defense in depth and, like the structural fire barriers in which they are installed, are passive fire-protection features. Their design function is to confine a fire to the area in which it started and to protect plant systems and components within an area from a fire outside the area.

Between 1994 and 1996, the NRC staff sponsored a comprehensive technical assessment of penetration seals to address reports of potential problems (e.g., aging), to determine whether any significant safety problems existed, and to determine whether NRC requirements, review guidance, and inspection procedures were adequate. The staff did not find plant-specific problems of safety significance or any concerns with generic implications. Therefore, the staff concluded that the general condition of the industry's penetration seal programs was satisfactory.[9]

Turbine Building Fire Hazards

Before the turbine overspeed and fire event at Salem Unit 2, on November 9, 1991, the NRC's concerns about turbine hazards were primarily focused on large, high-energy missiles. However, the Salem event demonstrated that discharges of hydrogen and lubrication oil during turbine overspeed events could result in explosions and fires. These disruptions could also create reactor protection challenges, such as reactor trips, losses of main feedwater with demands for ECCS actuation, main steam line cracks or breaks as a result of excessive turbine vibrations, or instabilities of offsite power sources resulting from generator or main transformer faults.

An NRC assessment of those concerns at several plants indicated that safe shutdown could still be achieved because safety-related shutdown systems were separated from turbine building fire hazards

[9] NUREG-1552, "Fire Barrier Penetration Seals in Nuclear Power Plants," Supplement 1, issued January 1999.

by 3 hour fire barriers. As a result, the probability of a major turbine fire adversely affecting those systems was low. However, the smoke, fire-fighting activities, or failure of fire barrier components during such fires could pose residual risks to safety-related shutdown systems. Therefore, the assessment indicated that plant-specific fire PRAs—such as the IPEEE discussed at the end of this Chapter—should consider these risks.

Overall, the two fire analysis zones found most often by the IPEEEs to be the highest fire CDF contributors were switchgear areas and MCRs. The third most commonly identified area was the turbine building.[10] However, the only plants that identified "fire vulnerabilities" in their IPEEEs found they were caused by turbine building fires. Thus, the IPEEEs indicated that any plant-specific fire PRA should consider the risk from turbine building fires.

Operational Data: Office of Analysis and Evaluation of Operational Data Study

A comparison of fire events in the pre-Appendix R period (1965–1985) with fire events in the subsequent period (1986–1994) showed that their frequency declined slightly and their safety significance was lower in the latter period. As previously discussed, in the former period, the most significant fire event occurred at BFN. This fire, which resulted in reactor shutdown and fire propagation without suppression, damaged multiple redundant trains of safety equipment. Also in that former period, 10 other fire events resulted in reactor shutdown with loss of one safety-related train or loss of offsite power (LOOP). In the latter period, no fire events occurred with safety significance comparable to the BFN fire, and only two LOOP events resulted. Although additional fires occurred during the latter periods that were severe in terms of their magnitude and duration of combustion (such as turbine building fires), their severity was limited in terms of challenges to safety systems. However, they could have been significant if redundant safety trains had been dependent on equipment located in the same fire area.

The industry and the NRC continue to support the analysis of data on fire events in order to remain alert to any trends in NPP fires (e.g., in their overall frequency, severity, type, or location). One project, begun in 2008, is the creation of a "consolidated fire events database" from the several NPP fire databases that currently exist (as mentioned in Chapter 7).

Electric Circuit Faults (Hot Shorts) and Self-Induced Station Blackout

A fire-related "hot short" is a conductor-to-conductor short circuit in which one or more nonenergized, nongrounded conductors become energized because fire damage has caused cable insulation failure. As shown by experience (e.g., the fire at BFN) and experiments (e.g., those discussed in Chapter 6, in the "Tools for Circuit Failure Mode and Likelihood Analysis" section), a hot short can cause the spurious operation of plant equipment such as pumps and motor-operated valves, resulting in possible damage to, or the defeat of, safety-related shutdown systems. A hot short can also cause erroneous instrument readings, which can lead plant operators to take inappropriate actions. Both spurious operations and inappropriate operator actions can significantly increase the probability of core damage caused by fire.

The NRC's fire-protection regulations recognize that fires can induce multiple hot shorts and require that the plant nevertheless be capable of achieving safe shutdown. Those regulations require adequate separation of redundant trains of safety-related equipment and adequate procedures

[10] NUREG-1742, "Perspectives Gained from the IPEEE Program," Volumes 1 and 2, April 2002.

for their use to safely shut down the plant in the event of fire. But for fires in areas where additional risk reduction is desirable, such as the MCR, safe shutdown is accomplished by employing alternative shutdown procedures that direct operators to carry out shutdown activities at control stations located outside the MCR (e.g., at the remote shutdown panel (RSP)).

To protect the equipment operated at the RSP from fire-induced electrical problems, particularly spurious actuations, plants often use electrical isolation switches that are actuated after the MCR is evacuated. Plants employing this design are said to have an electrical isolation scheme (EIS). A few plants without an EIS implement some form of temporary, deliberate bus deenergization (e.g., self-induced station blackout (SISBO)) to prevent these spurious actuations.

The NRC staff was concerned that certain safe-shutdown-related equipment and components could be damaged by fire-induced faults before electrical transfer and isolation (by the EIS or SISBO) could be accomplished. The staff also was concerned that the additional risks incurred by deliberately placing the plant in a more degraded condition (i.e., SISBO) might be unnecessarily large. Similar issues were considered in GI 147, "Fire-Induced Alternate Shutdown Control Room Panel Interactions," which focused on the potential for a control room fire to induce multiple spurious actuations and equipment failures before control was transferred to the remote shutdown station. GI 147 showed that identification of such potential multiple failure vulnerabilities would require the plant-specific evaluation of equipment fault states using a probabilistic approach.

The NRC staff continued to evaluate these concerns, first, through review of the IPEEEs, discussed at the end of this chapter, and later, using several fire-test programs and expert panel elicitations as part of the fire-risk quantification process discussed throughout this document, especially in Chapter 6. Such evaluations continued to demonstrate that the analyses of risk from fires at NPPs must use a probabilistic approach that includes consideration of (potentially multiple) hot-short-related equipment failures.

Certain Generic Issues

Although they were not specifically conducted as part of its fire research program, the NRC identified the following five GIs (in addition to GI 57, GI 147, and GI 149 discussed above) that are associated with fire safety and that could potentially influence future PRA improvements.

(1) **GI 81**—Degraded Access Caused by Locked Doorways. This issue involves the conflicting needs for access control to provide plant security and for unimpeded access for quick emergency response. In terms of fire safety, emergency responses that could be compromised by access control constraints are manual fire response and operator control and recovery operations. The balance between these conflicting needs is difficult to achieve and depends heavily on the particular plant design, which determines the areas requiring unimpeded access in the event of an emergency. Research conducted on this issue determined that the mechanical key overrides present on most vital area doors (along with the issuance of keys to operations personnel), plus the ability of operators to quickly obtain tools and physically force the doors open if necessary, were sufficient to reduce risk to acceptable levels. However, the risk is highly plant-specific, and an evaluation of a plant's locked doors and barriers might be required to establish that operator access is unimpeded under emergency, abnormal, or accident conditions, and that prompt and vital operator actions are possible.

(2) **GI 83**—Control Room Habitability. The NRC developed a number of requirements for the design of ventilation systems for control rooms to ensure their habitability in the event of a plant accident. These requirements include

protection of ventilation system components and power sources, the ability to isolate control rooms from all external inputs, and the provision of emergency air supplies for use by plant operators. However, for certain fires, tests[11] have demonstrated that achieving effective smoke removal from control rooms is very difficult, even at high ventilation rates. Thus, procedures are also needed for the abandonment of MCRs in the event of an uncontrolled fire. In addition, emergency air supplies (including eye irritation protection) should be available for all plant operating staff in NPP control rooms. Each licensee was required to confirm[12] that its facility's control room met applicable habitability requirements and that its habitability systems were designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing bases.

(3) **GI 106—**Piping and Use of Combustible Gases in Vital Areas. A potential source of fires in NPPs is leaks in flammable gas lines and systems (e.g., hydrogen). For light-water reactors, hydrogen is used as a component cooling medium (e.g., in the main generator) and in the reactor chemistry maintenance system. Many of the hydrogen lines are located in the auxiliary building, which is a safety-related structure housing many components of the plant's safety-related systems. A major part of the research for this GI was devoted to evaluating the costs and benefits of installing excess-flow check valves close to hydrogen storage tanks. The cost/benefit results were in a range that did not allow such installations to be generically required at all plants. Nevertheless, a licensee who has flammable gas piping in vital areas should consider including the issue in its PRA.

(4) **GI 107**—Transformer Failures. NPP operating experience includes numerous transformer failures that resulted in severe fires; many resulted in significant operational challenges to safety, and some spread beyond the transformer area. This GI assessed the high failure frequency of main transformers and their resultant safety implications. It determined whether licensees should be required to perform the following actions: evaluate the main transformer design and arrangement to ensure that the supply of offsite power would be protected against transformer fires and smoke; review fire protection system features for main transformers to ensure that a fire would not spread to other plant areas; review maintenance and operating procedures for main transformers to reduce the frequency of fires as much as possible; consider modifying transformer drainage systems so flammable liquids would flow away from the turbine building, power lines, and safety-related cables; consider modifying fire-fighting equipment and procedures (e.g., to include longer hoses, increase the ease of access to building roofs, increase the mobility of fire-fighting equipment); and consider relocating power lines that supply the safety-related buses so they would not be affected by a fire in the transformer bay. Although these actions were in a cost/benefit range that did not allow them to be required, licensees should consider the transformer fire issue for inclusion in PRAs, especially for plants with features that might make them particularly vulnerable.

(5) **GI 148**—Smoke Control and Manual Fire Fighting Effectiveness. This issue focused on the high reliance most NPPs place on the ability of onsite fire-response teams to rapidly suppress fires, which requires a high level of training for those personnel. However, the NRC requires only a minimum level of staffing and training, which may result in no live-fire training for the fire-response teams. A review of

[11] NUREG/CR-4681, and NUREG/CR-4527, Volumes 1 and 2.

[12] NRC Generic Letter (GL) 2003 01, "Control Room Habitability," dated June 12, 2003.

plant fire protection practices found that plants varied widely, with many plants reporting compliance with only the minimum standards.[13] Testing[14] has demonstrated that typical NPP fires can be expected to develop rapidly, creating a thick toxic smoke layer within the fire enclosure. To ensure that fire-response teams are adequately prepared to deal with such fires, training in excess of the minimum requirements has been recommended, including live-fire and smoke-room training.[15] In addition, at least one member of the fire-response teams for each shift must have the equivalent of operator-level knowledge of the plant systems and operational needs and procedures. It is vital for the teams to recognize the operational importance of given fire areas and the components in those areas to minimize significant collateral damage from fire suppression efforts.

Review of the IPEEE Fire PRAs

On June 28, 1991, the NRC issued Supplement 4 to GL 88 20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated September 8, 1995. Specifically, the NRC requested that each licensee perform an IPEEE to identify and report all plant-specific vulnerabilities to severe accidents caused by external events, including fires. All licensees completed and submitted their analyses to the NRC for review. The review determined whether the licensee's process was "capable of identifying severe accident vulnerabilities to such external events and implementing cost-effective safety improvements to either eliminate or reduce the impact of those vulnerabilities. However, the reviews did not attempt to validate or verify the licensees' IPEEE results."

Out of all 70 IPEEE submittals, only two licensees (representing three NPP units) identified significant fire vulnerabilities.[16] For both, the vulnerabilities included fire-safety issues in the turbine buildings, which housed important safety-related cables and equipment needed for safe shutdown. The turbine buildings also contained substantial fire sources, and in both cases, postulating large fires in the turbine buildings led to identifying the fire vulnerabilities. Many other licensees also identified turbine building areas as important CDF contributors.

Although the vast majority of licensees did not identify significant fire-related vulnerabilities, "over 60 percent" (i.e., over 42 licensees) did identify or implement plant improvements to reduce fire risk—for a total of approximately 240 fire-related plant improvements. The majority of the cited plant improvements ("about 60 percent") were associated with various plant procedures (e.g., operating procedures, maintenance procedures, combustible controls, enhanced operator training, enhanced fire brigade training). The remaining improvements involved plant modifications and hardware changes (general plant system design changes, enhanced fire-protection features, relocation of critical cables, and upgraded fire barriers).

The CDF values reported in the IPEEE fire analyses nominally confirmed that fire can be a potentially important contributor to overall plant CDF. In the vast majority of cases, licensees concluded that the dominant fire CDF contributors were areas that held both significant fire sources and important equipment and cables. Hence, it appears that spatial factors (e.g., the location of the fire source and targets) were more significant in determining fire risk than were plant system design features.

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13 NUREG/CR-5384.

14 Documented in NUREG/CR-4681, and in NUREG/CR-4527, Volumes 1 and 2.

15 SAND90-1827, p. 17, Steven P. Nowlen, SNL.

16 One of the licensees later identified additional equipment and operator recovery actions that should have been credited (which negated the "vulnerability") but nevertheless made plant improvements.

Chapter 6: Fire Research Initiated After the Commission's Policy Statement Encouraging Use of Probabilistic Risk Analysis Methods in Regulatory Matters, 1998–2008

This fire history summarizes the NRC-sponsored research to identify and reduce fire-related challenges to the ability of an NPP to achieve safe shutdown, thereby preventing nuclear core damage. It discusses many topics related to causes of fire risk and what has been done in the past (and is planned for the future) to lessen that risk. The topics slowly evolve as the history proceeds, changing from the use of deterministic, qualitative research methods to more quantitative PRA-informed methods. As stated in this history's Preface, the NRC formalized this trend toward the increased use of quantitative methods in NPP regulations in 1995 by issuing the following Commissioners' policy statement:

> The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

To support that policy, in 1998, the RES staff initiated a new part of its fire research program to address gaps in the ability to perform realistic fire PRAs. The NRC planned this program so that it would make the most efficient use of available resources to fill the most risk-significant gaps. Therefore, the first step in planning the program was to identify and prioritize those gaps.

Accordingly, those persons and organizations knowledgeable in relevant disciplines (e.g., PRA, fire-risk analysis, fire protection, fire safety) who contributed information included other NRC offices (e.g., NRR); other Government agencies (e.g., the National Institute of Standards and Technology (NIST)); certain national laboratories (e.g., BNL and SNL); universities (e.g., the University of Maryland); nuclear industry groups (e.g., the Nuclear Energy Institute [NEI] and EPRI); and several NPP licensees.

After several iterations, the NRC used the collected information to select the initial specific tasks for this new part of the fire research program, four of which were included in a DOE contract with SNL in June 1998. The four tasks were:

(1) "Tools for Circuit Failure Mode and Likelihood Analysis." This task included a worldwide literature search for "hot short" and resulting circuit failure data, which led to SNL's participation in industry-sponsored, full-scale fire tests that later supported an expert elicitation panel's interpretation of the test data for use in fire PRAs.

(2) "Tools for Fire Detection and Suppression Analysis." This task collected data used to improve methods to determine the likelihood that a fire will be detected and suppressed before it can damage critical equipment.

(3) "Fire Modeling Toolbox: Input Data and Assessment." This task collected basic data needed to assess the flammability and damageability characteristics of critical equipment in fires and the validity of physical models then available for predicting the fire-induced environment.

(4) "Experience from Major Fires." This task included a worldwide search for information related to three categories of significant fires in NPPs: (1) large or severe fires, (2) fires that led to a significant challenge to nuclear safety, and (3) "interesting" fires (i.e., those that involved an unusual chain of events or phenomenon).

During this time, a separate task was performed by Buttonwood Consulting, Inc., that focused on fires that are potentially challenging from a risk point of view, i.e., fires that can challenge risk-important targets within the plant. It proposed a mechanistic model that can calculate the frequency of those fires by linking fire initiation and subsequent fire modeling, and demonstrated that the initial stages of a variety of challenging, real-world fires fit the proposed model.

Upon completion of the initial tasks of this fire research program, the resulting new or improved data and methods would be tested and modified as indicated by trial applications to fire PRAs that had previously been performed for one or more NPPs; this task was referred to as the "requantification" of those fire PRAs. As work progressed on the initial tasks, it was decided that the requantification effort would be merged with work then underway by EPRI to improve fire PRA methods and data. Under the provisions of a Memorandum of Understanding (MOU), EPRI agreed to cooperatively undertake this combined requantification.

Because the SNL contract and the development of the joint NRC/EPRI fire-risk methodology described in NUREG/CR-6850 (EPRI 1011989) both involved the same SNL personnel, the results of the initial SNL tasks were seamlessly incorporated, where appropriate, into the joint methodology.

Two licensees partially requantified their fire PRAs using the joint fire-risk methodology. Unfortunately, neither licensee was able to participate in the pilot program as fully as intended, so the methodology has not yet been fully validated by trial and potential modification (i.e., the planned requantification effort has not been completed). However, it will be completed as described in Chapter 7, Item 1.

The following sections discuss these four tasks and the joint NRC RES/EPRI NUREG/CR-6850 (EPRI 1011989) fire PRA methodology, as well as other tasks, designed to further improve the ability of the NRC and the nuclear industry to perform realistic quantitative fire-risk assessments.

Tools for Circuit Failure Mode and Likelihood Analysis

This task developed a method to resolve the previously discussed conductor-to-conductor "hot short" issue. It began with a review of worldwide fire data related to electric cable failure modes and the likelihood of their occurrence. Its investigators then participated in a fire-testing program to enhance that data in conjunction with a program sponsored by EPRI and NEI in January–May 2001.[17] Next, its investigators participated in a joint NRC/SNL/EPRI expert elicitation panel that considered the data and recommended probabilities for fire damage and conditional probabilities for spurious actuation of electric devices, assuming specific combinations of cable types, time-temperature variations, and fire-severity conditions. The investigators used the data and expert panel recommendations to develop a proposed framework to include cable and circuit failure modes and likelihood analyses in an improved fire PRA method. The NRC published details of the data search, fire testing, and improved PRA method in NUREG/CR-6834, "Circuit Analysis—Failure Mode and Likelihood Analysis," issued September 2003.

The staff uses the insights and recommendations in NUREG/CR-6834 when reviewing fire-risk assessments and when updating the NRC's research plans. The NRC has broadly distributed the report and has made it publicly available on its Web site, because its detailed discussions of available data (and methods to use in applying those data) will be useful to anyone performing a fire-risk analysis.

The joint NRC/EPRI fire-risk methodology described in NUREG/CR-6850 (EPRI 1011989) cites the above-mentioned fire-testing program

[17] NUREG/CR-6776, "Cable Insulation Resistance Measurements Made During Cable Fire Tests," issued June 2002, contains details of the NRC-sponsored results.

and expert elicitation as major contributors to the "estimating spurious actuation probabilities" portion of the methodology.

Tools for Fire Detection and Suppression Analysis

A fire PRA requires a method for analyzing fire detection and suppression to determine the likelihood that a given fire will be suppressed before it damages critical safety equipment. This task was to provide an improved fire detection and suppression modeling framework and data for estimating the reliability of automatic and manual suppression activities; to develop estimates of these reliabilities for currently operating nuclear plants; and to identify and quantify key uncertainties in these estimates.

One specific recommendation from this task was to assess the fire brigade response based on scenario-specific brigade response times and historical evidence regarding fire suppression times for similar fires. Accordingly, fire suppression time curves were derived for various groupings of specific fire-ignition sources, fire locations, and fire types, based on an analysis of the EPRI fire-event database.

The results of this task were considered in developing the fire detection and suppression method recommended in NUREG/CR-6850 (EPRI 1011989) that includes the use of historical evidence regarding fire duration gleaned from fire events, combined with the use of fire brigade response times demonstrated by unannounced fire drills.

Fire Modeling Toolbox: Input Data and Assessment

The reliable prediction of the hazardous environment induced by a fire and the response of critical equipment to that environment are important parts of a fire PRA. Key uncertainties in these predictions are caused by the sparseness of basic data needed to assess the flammability and damageability characteristics of equipment under fire conditions and the validity of available physical models for predicting the fire-induced environment.

A major purpose of this task was to collect and characterize experimental data from previously performed experiments relevant to the above—in particular, to process raw experimental data collected under the Large-Scale Base Line Validation Test (documented in NUREG/CR-4681 and NUREG/CR-4527, Volumes 1, and 2). These tests were conducted in an enclosure measuring 60×40×20 feet, with forced ventilation systems designed to simulate typical NPP installation practices and ventilation rates. SNL conducted a total of 25 tests using gas burner, heptane pool, methanol pool, and solid fires, four of them with a full-scale control room mockup in place. The tests varied the parameters for fire intensity, enclosure ventilation rate, and fire location. Data included air temperatures, air velocities, radiative and convective heat flux levels, optical smoke densities, inner and outer enclosure surface temperatures, enclosure surface heat flux levels, and gas concentrations within the enclosure in the exhaust stream.

A secondary purpose of this task was to determine the appropriate value of the heat loss factor (HLF) that should be used in the simplified hot gas layer response model from the fire-induced vulnerability evaluation (FIVE) computer code used by the nuclear industry (EPRI). In particular, SNL assessed both the actual HLF characteristics of the "base line validation" data and the appropriate value of the HLF that should be assumed when using the FIVE computer code to predict test results.

The NRC considered the results of this task in developing the data and methods recommended in NUREG/CR-6850 (EPRI 1011989) for predicting hazardous environments induced by fires and the response of critical equipment to those environments.

Experience from Major Fires

This task was to gain new methodology insights from actual NPP fire incidents worldwide. The study reviewed 25 fire incidents, including fires at both U.S. and foreign reactors, selected in three categories: (1) large or severe fires, (2) fires that led to a significant challenge to nuclear safety, and (3) "interesting" fires (i.e., those that involved an unusual chain of events or phenomenon). The sequence of actions and events observed in each fire incident was reconstructed based on available information. This chain of events was then examined and compared to typical assumptions and practices of fire PRAs. The review focused on two types of actions and events: (1) events that illustrate interesting insights regarding factors that fell within the scope of current methods, and (2) events observed in actual fire incidents that fell outside the scope of current methods.

The review concluded that the overall structure of current fire PRA methods can appropriately capture the dominant factors involved in a fire incident. However, the review identified several areas of potential improvement to current methods. One improvement would more realistically consider the effects of smoke propagation on plant operators and fire fighters, which could lead to event sequences otherwise considered very unlikely. The review also identified a few factors that fell outside the scope of current fire PRAs, including the occurrence of multiple initial fires or secondary fires, multiple simultaneous initiating events, and turbine blade ejection events that could simultaneously result in fires caused by significant releases of flammable lubrication oil and hydrogen, mechanical damage from blade debris (such as severance of pipelines), and flooding (e.g., damage to water lines containing river water).

The NRC published the detailed results and insights in NUREG/CR-6738, "Risk Methods Insights Gained from Fire Incidents," in September 2001. The NRC staff believes the insights in NUREG/CR-6738 would be useful to anyone involved in fire-risk analysis. For example, as discussed in this document, the insights were considered by the joint NRC/SNL/EPRI group that developed the data and methods recommended in NUREG/CR-6850 (EPRI 1011989). The insights provided that group with confidence that no significant revisions were needed to the general fire PRA approach currently being used and that improvements could be incorporated through more readily accomplished changes to specific fire PRA elements.

Development of a Joint NRC/EPRI Fire Risk Methodology

A joint RES/SNL/EPRI group considered the results of the above tasks, certain similar efforts by the nuclear industry (EPRI), and current fire PRA data and methods in creating its recommended "consensus" fire PRA methodology for use at NPPs. NUREG/CR-6850 (EPRI 1011989) documents this method and notes that its scope "is limited to fire risk during at-power mode of operation."

Participants from the U.S. nuclear power industry supported demonstration analyses and provided peer review of this methodology. The project addressed methodological issues raised in past fire-risk analyses, including the IPEEE fire analyses, to the extent allowed by the state of the art at that time and the overall project scope. Although the primary objective of the project was to consolidate existing state-of-the-art methods, in many areas, the newly documented methods represented a significant advance over previously documented methods. The project also provided the basis for the joint ASME/ANS RA Sa 2009 Standard, "Addenda to ASME/ANS RA S 2008: Standard for Level 1/Large Facility Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," issued in 2009, that includes fire PRAs. As previously noted, the demonstration/trial of this method will be completed as described in Chapter 7, Item 1, of this document.

Additional Support for Fire PRA Improvements Following the Initial Tasks

The most significant research efforts in this group fall into the five areas discussed below: (1) improving fire modeling, (2) obtaining additional data to reduce uncertainty related to hot shorts and circuit failures, (3) determining the qualification and performance of certain fire-barrier materials, (4) developing improved methods to account for human performance under the psychological stresses caused by fires, and (5) ranking the safety importance of key phenomena associated with the intended applications of fire models. These five tasks are further discussed below.

Improving Fire Modeling—Verification and Validation

Two outgrowths of the 1995 Commissioners' policy statement that "…the use of PRA technology should be increased in all regulatory matters…" were: (1) the National Fire Protection Association (NFPA) publication in 2002 of NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" (2001 Edition); and (2) the NRC's subsequent July 2004 amendment of 10 CFR 50.48, "Fire Protection," to permit existing reactor licensees to voluntarily adopt the fire-protection requirements contained in NFPA 805 as an alternative to existing deterministic fire-protection requirements. In addition, the NPP fire-protection community has been using risk-informed, performance-based (RI/PB) approaches and insights (that use fire PRAs) to support fire-protection decision-making in general.

One key tool needed to further the use of RI/PB fire-protection methods is the availability of verified and validated fire models that can reliably predict the consequences of fires. NFPA 805 states that only fire models acceptable to the Authority Having Jurisdiction (in this case, the NRC) shall be used in fire-modeling calculations. Moreover, NFPA 805 states that fire models shall only be applied within the limitations of the given model and shall be verified and validated.

This task was a joint RES/EPRI documentation of the verification and validation (V&V)[18] of five fire models used in NPP applications. The results of this V&V are reported in the form of ranges of accuracies for the fire-model predictions in the seven-volume "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," NUREG-1824 (EPRI 1011999), issued May 2007. These reports were produced under an interagency MOU with the U.S. Department of Commerce, NIST. Since the inception of efforts to provide risk-informed fire-safety analyses, the NRC has collaborated with NIST; this has resulted in significant contributions to those efforts, including contributions to this fire model V&V project.

Additional Data to Reduce Uncertainty Related to Hot Shorts and Circuit Failures (and to support fire model improvements)

As mentioned in the previous section, "Tools for Circuit Failure Mode and Likelihood Analysis," to better understand the issue of cable hot shorts, the nuclear industry (NEI/EPRI) conducted a series of cable fire tests in 2001. Based on the results of those tests and data from previous tests available in the literature, the NRC facilitated a workshop on February 19, 2003, to produce guidance for NRC inspectors to follow in determining which causes of fire-induced hot shorts were important to safety and should be considered during inspections. The

[18] Verification is the process used to determine that a model correctly represents the developer's conceptual description. It is used to decide whether the conceptual model was "built" correctly. Validation is the process used to determine that a model is a suitable representation of the real world and is capable of reproducing phenomena of interest. It is used to decide whether the right model was "built."

workshop led to the issuance of Regulatory Issue Summary 2004 03, Revision 1, "Risk-Informed Approach for Post-Fire Safe Shutdown Circuit Inspections," dated December 29, 2004, which also describes scenarios where the importance to safety of cable hot shorts was unknown at the time of the workshop (these scenarios were referred to as "Bin 2" items).

Primarily to determine the safety importance of these Bin 2 cable hot-short issues, SNL conducted the CAROLFIRE (Cable Response to Live Fire) testing program. CAROLFIRE also made thermal measurements to support the secondary objective of supporting fire-model improvements (i.e., in addition to electrical resistance measurements to determine electrical performance, thermal measurements were made to support fire modeling).

CAROLFIRE included a series of both small- and intermediate-scale cable fire tests comprising 96 individual experiments of varying complexity and involving a variety of common ca-ble constructions. SNL also changed various test parameters from test to test, including thermal exposure, raceway type, and bundling of similar and dissimilar cable types. The CAROLFIRE results are the most extensive set of combined electrical and thermal response data currently available. They provide valuable information and insights that will improve the treatment of cable failures and the resulting hot shorts and circuit malfunctions (e.g., spurious actuations) in fire PRA methods.[19] NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," issued April 2008. Volume 1: "Test Descriptions and Analysis of Circuit Response Data," documents the CAROLFIRE electrical performance data; Volume 2: "Cable Response Data for Fire Model Improvement," documents the thermal data. Figure 3 shows two cables in a 30.48 centimeter (cm) (12 inch (in.)) cable tray.

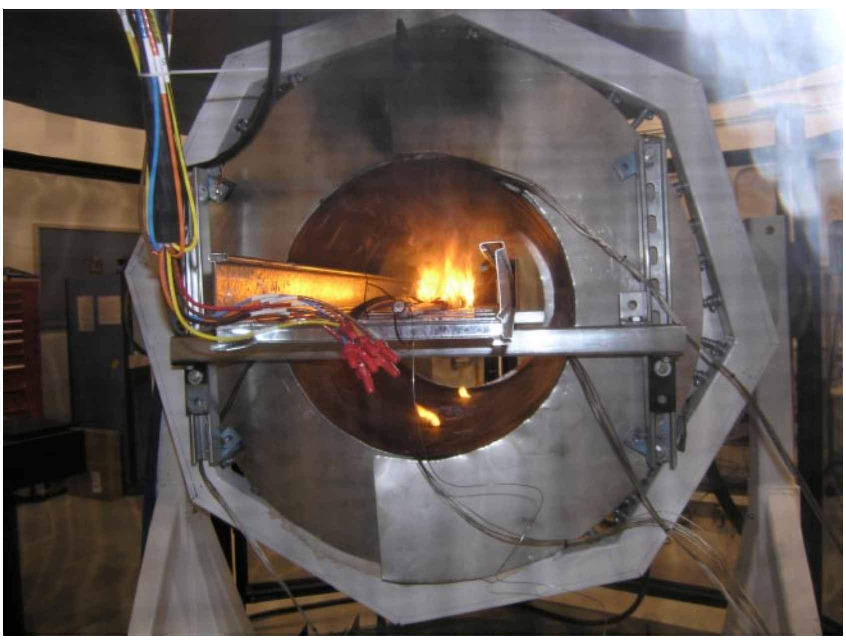

Figure 3 Small-scale CAROLFIRE test—two cables in a 30.48 cm (12 in.) cable tray

NIST used the data collected by SNL during CAROLFIRE's electrical and thermal performance tests to develop the Thermally-Induced Electrical Failure (THIEF) model. Incorporation of THIEF as a subroutine in a deterministic fire model will enable that model to use its thermal calculations to predict the electrical performance of cables in a fire. NUREG/CR-6931, Volume 3: "Thermally-Induced Electrical Failure (THIEF) Model," issued April 2008, documents the THIEF model.

[19] Three examples (R. Gallucci) are: bounding the number of concurrent spurious operations; probabilistic/statistical examination of cable hot short durations caused by NPP fires; and probability of fire-induced cable failure as a function of exposure temperature and time. These papers are cited only to provide sample applications of the data; this does not imply NRC endorsement or approval of the papers' results.

The Qualification and Performance of Certain Fire-Barrier Materials

Although this project did not involve quantitative safety evaluations, the issue is nevertheless included in this section for two reasons: (1) it was performed during the time period covered by this section and (2) final resolution of the compliance issue that resulted from the project will likely be based on quantitative safety analyses for some NPP licensees.

The effectiveness of fire barriers was previously mentioned in this document as one of the NPP features to be confirmed to ensure the validity of the new Appendix R fire-protection requirements after the BFN fire. Two of the materials used to construct such fire barriers are "Hemyc," a composite material consisting of a high-temperature-resistant Kaowool thermal insulation blanket core, usually 3.81 or 5.08 cm. (1 1/2 or 2 in.) thick, enclosed in an envelope of high-temperature resistant welding cloth (e.g., "Refrasil"), and "MT," a similar material with additional layers of metal sheeting and a material that absorbs heat by melting, thereby providing longer thermal protection. NPPs have used both materials to construct Electrical Raceway Fire Barrier Systems (ERFBS) to protect vital electrical cables from fire damage.

The two materials must perform so as to protect cables from exceeding specified temperatures for 60 minutes (Hemyc) or 180 minutes (MT) during exposure to specified standard fires (unlike other items in this section, these required performance levels were not based on quantitative safety evaluations). If such materials do not meet their required performance levels, a regulatory compliance issue results. The compliance issue must be resolved by replacing the materials with an acceptable material, by compensatory measures, or by a formal exemption from the requirements.

The origin of the compliance issue was a 2003 memorandum titled "Hemyc Fire Test Plan" from NRR, requesting RES to conduct a fire-endurance testing program to determine the fire- resistance capability of ERFBS constructed using documented, vendor-approved Hemyc and MT ERFBS materials and documented, vendor-approved installation techniques. By contract with SNL, RES conducted three full-scale fire tests: a 1 hour fire test for Hemyc ERFBS protecting conduits, a 1 hour fire test for Hemyc ERFBS protecting cable trays, and a 3 hour fire test for MT ERFBS protecting conduits.

The tested ERFBS all failed to meet their respective acceptance criteria—the 1 hour Hemyc ERFBS lasted between 15 and 43 minutes, and the 3 hour MT ERFBS lasted between 87 and 159 minutes. The tests failed because the outer covering materials thermally shrank, which generated forces that exceeded their strength and opened a gap at their weakest point. The weakest point was most often at a seam between two pieces of the material or at a fastening where the material was connected to the underlying electric raceway. If no seams existed or if the seams were exceptionally strong and connections to the raceway were made so the thermal shrinkage-generated forces were distributed over a large area, the outer covering material itself would rip.

The above test failures resulted in a compliance issue, which led NRR to issue Information Notice (IN) 2005 07, "Results of Hemyc Electrical Raceway Fire Barrier System Full-Scale Testing," dated April 1, 2005, and GL 2006 03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations," dated April 10, 2006. Figures 4 and 5 show a Hemyc Fire Barrier Test on cable trays (pretest and posttest, respectively).

Some licensees responded to the compliance issue by replacing the materials that failed with qualified materials. Other licensees, who are adopting NFPA 805 as an alternative to existing deterministic requirements, may be able to show that the existing performance of their Hemyc and MT ERFBS

Figure 4 Hemyc fire barrier test on cable trays, pretest

Figure 5 The same Hemyc fire barrier test on cable trays, posttest (note failed, glowing joints)

(e.g., as measured by the SNL tests) provides an acceptable level of safety.[20] This is an example of the advantage of risk-informed regulation; it could allow those licensees who are adopting NFPA 805 to avoid the expense of unnecessary modifications while still providing acceptable safety.

Improved Methods to Account for Human Performance in Fires

During the 1990s, EPRI developed methods for fire-risk analysis to support its utility members in the preparation of responses to GL 88 20, Supplement 4, "Individual Plant Examination—External Events" (the IPEEE, previously discussed). That effort produced a fire-risk assessment methodology that was used by the majority of NPPs in support of the IPEEE program. Although those methods were acceptable for accomplishing the objectives of the IPEEE, EPRI and the NRC recognized that those methods required upgrades to support current requirements for RI/PB applications.

In 2001, EPRI and RES embarked on a cooperative project to improve the state of the art in fire-risk assessment to support the new risk-informed environment in fire protection. That project produced the previously discussed joint NRC RES/EPRI NUREG/CR-6850 (EPRI 1011989) document that addresses fire risk. That report developed (1) a process for identification and inclusion of post fire human failure events (HFEs), (2) a methodology for assigning quantitative screening values to these HFEs, and (3) the initial considerations of performance-shaping factors and

[20] Hemyc paper by R. Gallucci. This paper is cited only to provide an example of this possibility; this does not imply NRC endorsement or approval of the paper's results.

related fire effects that may need to be addressed in developing best-estimate human error probabilities (HEPs). However, that document did not describe a methodology to develop these best-estimate HEPs, given the performance-shaping factors and the fire-related effects.

In 2007, EPRI and RES embarked on another cooperative project to develop explicit guidance for estimating HEPs for HFEs under fire-generated conditions, building on existing human reliability analysis (HRA) methods. Current progress regarding development and testing of that fire HRA methodology includes addressing the range of fire procedures used in existing plants, the range of strategies for MCR abandonment, and the potential impact of fire-induced spurious electrical effects on crew performance. In addition to developing a detailed HRA approach, one goal of the project is to develop a fire HRA scoping quantification approach that allows derivation of more realistic HEPs than those in the screening approach in NUREG/CR-6850 (EPRI 1011989), while requiring fewer analytical resources than a detailed HRA. In this approach, a detailed HRA will be used only for the more complex actions that cannot meet the criteria for the scoping approach.

In October 2007, the NRC issued NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," which addresses the unique aspects of fire and associated operator manual actions. That report suggests criteria for use by the NRC staff in determining whether operator manual actions—proposed by operating plants for use in achieving and maintaining hot shutdown—are feasible and can be performed reliably in response to fire. Among the criteria are the existence of time-authenticated demonstrations of the manual actions (involving actual execution of the actions, to the extent possible) and the availability of adequate time to complete the actions before fire-induced consequences occur that would otherwise prevent achieving and maintaining hot shutdown.

Ranking Key Phenomena Associated with Fire Models

SNL facilitated a phenomena identification and ranking table (PIRT) exercise for NPP fire modeling applications using an expert elicitation process with an expert panel of seven internationally known fire-science experts. Its objective was to identify key phenomena associated with a fire model's intended application and then rank the current state of knowledge for each identified phenomenon. To do this, SNL presented the panel with a series of specific fire scenarios, each based on the types of scenarios typically considered in NPP applications. Each scenario included a specific goal to be achieved in its analysis, using fire-modeling tools (e.g., an MCR fire with the specific goal of predicting the time when the operator would abandon the control room).

For each scenario, the panel identified the phenomena that could affect the uncertainty in the predicted numerical value of the specific goal (e.g., time of control room abandonment in the above example). The identified phenomena were ranked relative to their importance in achieving the specific goal and were then further ranked, based on the existing state of knowledge and the adequacy of existing fire-modeling tools to predict each phenomenon. These results will give the NRC valuable technical insights into the predictive capabilities of fire-modeling tools. In addition, the NRC will use the PIRT results to identify areas where further research and analysis are needed. NUREG/CR-6978, "A Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire Modeling Applications," issued November 2008, presents the results of this exercise.

Chapter 7: Fire Research Underway at End of 2008

This history of NRC fire-safety research is not a complete story. A convenient time was chosen for it to begin (i.e., the creation of the NRC), but no convenient time was apparent for it to end, because the NRC and its fire research continues.

The most significant projects underway at the end of calendar year 2008 were:

(1) "Demonstration of the Joint NRC/EPRI Fire-Risk Methodology (NUREG/CR-6850)." The previous discussion of that methodology mentioned the plan to requantify one or more previously performed NPP fire PRAs, but that the demonstrations could not be completed because the NPP participants were not able to participate as fully as expected. This project will complete the demonstrations through use of the complete NUREG/CR-6850 method by the NFPA 805 pilot plants, Harris and Oconee.

(2) "Extension of the Joint NRC/EPRI Fire-Risk Methodology (NUREG/CR-6850) for Applicability to Low-Power and Shutdown Operations." The previous discussion of that methodology noted its limitation "to fire risk during at-power mode of operation." This project will create a modified and expanded method, based on the NUREG/CR-6850 method but applicable to low-power and shutdown operations.

(3) "Extension of Fire Testing to Direct Current Circuits." The CAROLFIRE tests, and the earlier EPRI/NEI tests in 2001, were largely confined to testing alternating current (ac) circuits. Very limited tests indicated that results using direct current (dc) circuits might be considerably different (e.g., especially with respect to spurious actuations). The "Direct Current Electrical Shorting In Response to Exposure-Fire" (DESIREE-FIRE) project will conduct similar tests for dc circuits.

(4) "Development of Fire Standards and Regulatory Guides (RGs)." This project will support the development of various standards and guides for fire-related risk quantification, such as the previously mentioned support that NUREG/CR-6850 (the "Fire PRA Methodology" document) provided for the ASME/ANS RA Sa 2009 Standard that includes fire PRAs. For example, RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009, provides generic support to the application-specific RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," issued May 2006, which in turn supports the use of NFPA 805, the risk-informed alternative to existing deterministic fire-protection requirements.

(5) "Documentation of the Results of Completing ERFBS Actions." The staff is preparing a NUREG report that will document completion and closure of this complex fire-barrier issue. The report will consolidate documentation regarding all known raceway fire-barrier systems, including their effectiveness, information regarding the fire-endurance testing of the systems, and the NRC basis for closing any related open issues. These systems include, but are not limited to, Thermo-Lag, Kaowool, Hemyc, and MT.

(6) "Fire-Model Users Guide." This is the final currently planned step in the NRC's ongoing preparations for more extensive use of fire models to support NRC's continuing transition to more risk-informed regulatory processes. The V&V of selected fire models, previously discussed, determined what scenarios can currently be analyzed by which fire models with acceptably small uncertainty. The PIRT exercise, also previously discussed,

indicated where further improvements should be considered in fire-modeling techniques. The thermal data acquisition portion of CAROLFIRE (Volume 2) discussed additional data to support fire model improvements, and the THIEF code's development by NIST using the CAROLFIRE data was presented. This "Users Guide" is the joint EPRI/NRC project to provide guidance for users of fire models on what can be accomplished with which fire models, and which methods are acceptable.

(7) "Cable Heat Release Rate (HRR) and Flame Spread." This project will perform fire tests on bundles of electrical cables to measure their HRR and flame-spread characteristics. This is needed because previous cable fire tests that were conducted to investigate these characteristics have fallen into two categories, each of which is limited with respect to use of its data for fire-modeling purposes. A variety of small-scale material characterization fire tests have been carried out, but until their results are more fully validated with data from larger-scale tests, caution must be exercised if they are used to predict full-scale fire behavior. In addition, large-scale fire qualification tests have been carried out, in which cable materials were qualitatively ranked on a comparative basis, but these tests typically did not provide useful data for fire-model calculations because their documentation did not adequately characterize the test fires. The quantitative data collected by this project will be used to develop more realistic cable fire models for fire PRAs, such as those using the methods of NUREG/CR-6850 in NFPA 805 applications.

(8) "Consolidated Fire-Events Database." Several databases for NPP fire events currently exist. This project will consolidate them into a single, coordinated and uniform database.

(9) In support of the NRC's Office of Nuclear Material Safety and Safeguards (NMSS), RES is currently performing two risk analyses—one related to a particular fuel recycling facility and the other related to spent-fuel storage units.

- *Red Oil Risk Analysis for a Fuel Recycling Facility*. In September 2000, the United States and Russia signed an agreement to reduce their respective stockpiles of surplus weapons-grade plutonium. Accordingly, DOE is sponsoring the construction of a mixed-oxide fuel fabrication facility (MFFF) to convert approximately 34 metric tons of the U.S. stockpile into MOX (mixed uranium and plutonium oxide) fuel that will be used in commercial NPPs, thereby effectively destroying the plutonium for weapons applications. The MFFF will use a liquid-liquid extraction (LLE) process. Worldwide operating experience with LLE facilities has shown that, under rare upset conditions, the LLE process can result in the inadvertent creation of a potentially explosive and flammable substance that has become known as "red oil." This project quantitatively evaluates the risk from a red oil explosion at the MFFF.

- *Evaluation of the Performance of Spent Nuclear Fuel Transportation Package Seals in Beyond-Design-Basis Fires*. This project will evaluate the performance of seals for spent nuclear fuel transportation packages (i.e., "shipping casks") in beyond-design-basis fires. This evaluation is needed to determine the extent to which the seals provide an additional barrier against radionuclide release (i.e., in addition to the fuel cladding) when subjected to temperatures that exceed the manufacturers' rated temperatures. In particular, it will determine the potential for degradation or failure of metallic or elastomeric seals over a wide range of temperatures. The NMSS staff and contractors have analyzed the thermal performance of several spent fuel nuclear transportation packages (for both road and rail transportation) in two real-world, beyond-design-basis tunnel fires and have concluded that, if a seal failure occurred, releases of radioactivity would be within regulatory dose limits. However, NMSS is also interested in confirming the actual performance envelope of seals for a transportation package subject to a beyond-design-basis fire (in lieu of conservatively assuming their failure). This information will be used to further develop risk insights related to the transportation of spent nuclear fuel.

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