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Thermal Barriers and Regulatory Frameworks: the U.S. Experience on Fire Protection

by

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I. Introduction

Good morning, ladies and gentlemen. I would like to begin by emphasizing how very pleased I am at the opportunity to be here, in this capacity, in this country, for this particular conference. This is my first visit to India as the Chairman of the U.S. Nuclear Regulatory Commission (NRC), and even within my first few days here, I have been reminded of certain invaluable cultural and moral benefits that have come about through the links between our two countries. Perhaps the most striking example came on Saturday, in my visit to the Gandhi Memorial in New Delhi, where I was reminded again of how much the U.S. civil rights movement--and in particular the convictions and principles of Dr. Martin Luther King, Jr.--benefited from the wisdom and example of non-violence (or more appropriately, “*ahimsa*”) as a force for positive political and social change, as embodied in the life of Mahatma Gandhi. Truly, this is only one example of the kind of rich and meaningful exchange that spans our two cultures--an impact that, in fact, has been of great personal significance to me.

During my visit here, in addition to my participation in this fire safety conference, I already have had the pleasure of meeting with Indian nuclear safety officials from both government and industry. I also will have the opportunity to visit several Indian nuclear facilities, including the Tarapur nuclear power plant. As many of you know, I have a fundamental belief in the value of cooperative international exchanges in the field of nuclear safety--whether those exchanges take the form of joint research, active dialogue, or simply sharing information on our experience and insights--to enhance our mutual understanding of regulatory approaches, technical challenges, risk perspectives, and safety-conscious solutions. Because the generation and regulation of nuclear energy and reactor byproduct materials are topics not confined by national boundaries, nuclear safety issues in one country can have a direct impact on how similar issues are viewed or treated in the nuclear power industry around the globe. Cooperative research efforts can result both in shared information and shared costs, thereby leveraging available research resources and enhancing research capabilities. Therefore, while we each must understand our own domestic issues, we also benefit by maintaining an awareness of the larger sphere of international energy demands, regulatory activities, and safety issues.

Fire safety clearly is such an issue--a topic in which not only India and the U.S., but also other countries have benefited by international exchanges of information. Today I would like to focus on an overview of the U.S. experience in fire protection. I will begin by providing an historical perspective on the origins and development of U.S. fire protection policy for commercial nuclear facilities. Next, I will discuss the incorporation of risk assessment into the regulation of fire protection. I will conclude with an overview of current Commission deliberations regarding the best way to proceed in regulating fire protection at U.S. nuclear power plants.

II. Genesis and Development of U.S. Fire Protection Policy

During the early days of the U.S. nuclear program, regulatory acceptance of the fire protection program at a given nuclear power plant was based on the broad performance objectives of one of the general design criteria found in the U.S. Code of Federal Regulations. These design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. The general design criterion for fire protection simply specifies: (1) that structures, systems, and components important to safety be designed and located to minimize the probability and adverse effects of fires and explosions; (2) that noncombustible and heat-resistant materials be used wherever practical; and (3) that fire detection and suppression systems be provided to minimize the adverse effects of fires on structures, systems, and components important to safety. However, during this early stage of nuclear power regulation, given the lack of detailed implementation guidance for this general design criterion, the level of protection provided at a given plant generally was found to be acceptable if the facility complied with local fire codes and received an acceptable rating from its fire insurance underwriter. In other words, the fire protection features installed in early U.S. nuclear power plants were very similar to those installed in conventional fossil-fuel power generation stations.

The Browns Ferry Nuclear Power Plant fire, on March 22, 1975, was a pivotal event that brought fundamental change to fire protection and its regulation in the U.S. The fire started when plant workers in the cable spreading room used an open flame to test for air leakage through a non-fire-rated penetration seal that led to the reactor building. The fire ignited both the seal and the cables that passed through it, and burned for almost 7 hours before it was put out using a water hose stream. The greatest amount of fire damage actually occurred on the reactor building side of the penetration, in an area roughly 40 feet by 20 feet. More than 1600 cables, routed in 117 conduits and 26 cable trays, were affected--and of those cables affected, over 600 were safety-related. The electrical cables, after insulation had burned off, shorted together and grounded to their supporting trays or to the conduits, with the result that control power was lost for most of the related equipment such as valves, pumps, and blowers. The fire damage to electrical power and control systems impeded the functioning of both normal and standby reactor cooling systems, and degraded plant monitoring capability for the operators. Given the loss of multiple safety systems, operators had to initiate emergency repairs in order to restore the systems needed to place the reactor in a safe shutdown condition.

The investigations that followed this event found significant inadequacies both in the design of fire protection features and in licensee procedures. The investigators concluded that the occupant safety and property protection concerns of fire insurance underwriters did not sufficiently encompass nuclear safety issues, especially in terms of the potential for fire damage to cause the failure of redundant trains of systems and components important for safe reactor shutdown. The NRC Browns Ferry special review team recommended that the NRC (1) should develop detailed guidance for implementing the general design criterion for fire protection; and (2) should conduct a detailed review of the fire protection program at each operating nuclear power plant, comparing it to the guidance developed.

The years that followed the Browns Ferry fire illustrated both the difficulties and the rewards associated with retroactively imposing new safety provisions on existing nuclear power plants. In May 1976, the NRC issued two sets of technical guidelines on fire protection, which incorporated the recommendations from the Browns Ferry fire special review team. One set of guidelines applied only to those licensees that filed for a construction permit after July 1, 1976. The second set, which applied to the, then, currently operating plants, established a less stringent, minimum level of fire protection intended to minimize the impacts on existing plant design and operation, while still maximizing the availability of plant equipment needed to achieve and maintain safe shutdown in the event of a fire. By the late 1970s, most operating plants had implemented most of the fire protection program recommendations specified, and in most cases, the NRC had found the licensee modifications to be acceptable. In certain instances, however, disagreements between a given licensee and the NRC staff led to licensee refusal to adopt some of the specified fire protection recommendations.

After considerable deliberation, the Commission determined that, given the generic nature of some of the disputed issues, a rulemaking was necessary to ensure proper implementation of NRC fire protection standards. Therefore, in November 1980, the NRC published its “Fire Protection” rule, which specified broad performance requirements, as well as Appendix R, which specified detailed regulatory requirements for resolving the disputed issues. Interestingly enough, the NRC staff originally intended Appendix R to focus only on resolving the previously disputed fire protection issues. During the rulemaking process, however, the Commission determined that certain Appendix R provisions (related to emergency lighting, fire protection of safe shutdown capability, and the collection of oil from reactor coolant pumps) were of such safety significance that they would be applied to all plants--even to those plants at which the NRC previously had approved the relevant fire protection features.

This overall historical progression--the broad provisions of the general design criterion, the detailed implementing guidance, the plant-by-plant review, and finally the issuance and backfit of the fire protection regulation and the prescriptive requirements of Appendix R--resulted in a complex regulatory framework for fire protection in U.S. nuclear power plants licensed prior to 1979. As an additional element of complexity, the new fire protection regulation specifically allowed a licensee to apply for an exemption to an Appendix R requirement, in cases where a fire hazard analysis could demonstrate that alternative fire protection features would provide an equivalent level of fire safety. During the initial backfit of the fire protection regulation, the NRC approved a large number of configuration-specific exemptions (i.e., alternative methods to achieve the underlying purpose of the regulation) at about 60 nuclear power plants. Exemptions were not automatic, and in many cases the NRC denied licensee exemption requests. In addition, since the mid-1980s, when licensees achieved compliance with the new fire protection regulations, the number of exemptions requested and approved has slowed to a trickle. Even so, the ongoing review of licensee fire protection programs, the licensee efforts to save costs while maintaining an acceptable level of safety, and the emergence of additional technical issues (such as the deliberations over the adequacy of Thermo-Lag as a fire protection barrier) have resulted in an overall total of 850 exemptions to specific elements of the NRC fire protection requirements. Note that for plants licensed <u>after</u> January 1, 1979, the provisions of Appendix R and the fire protection guidance were incorporated into the standard review plan used to review the fire protection program during the NRC licensing process.

III. The Incorporation of Risk Assessment Into the Regulation of Fire Protection

As convoluted as this preceding history may seem, the NRC regulatory framework for fire protection has been based on a philosophy of “defense-in-depth”: providing a balance among (1) preventing fires from starting; (2) rapidly detecting, suppressing, extinguishing, and limiting the spread of those fires that occur; and (3) designing safety systems so that, even if a fire starts and burns for a considerable time, it will not interfere with the performance of essential plant safety features. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a the nuclear facility. In recent years, the increasing sophistication of probabilistic risk assessment (PRA) methods have provided the potential for enhancing this traditional approach through the incorporation of risk insights.

In June 1991, the NRC requested U.S. nuclear power reactor licensees to perform a risk assessment to identify any vulnerabilities to severe accidents resulting from internal fires, to identify any low-cost improvements to mitigate the vulnerabilities, and to report the results of this analysis to the Commission. To perform the fire analysis, licensees had the option of using a Level 1 (core damage) PRA, the “Fire Induced Vulnerability Evaluation” (FIVE) developed by the Electric Power Research Institute (EPRI), or another systematic examination method acceptable to the NRC staff. Regardless of the method selected, this assessment was to identify vulnerable conditions that might not have been addressed fully by the fire protection regulation. Such conditions included: (1) seismic/fire interactions; (2) the effects of fire suppressants on electrical equipment; (3) the interactions of control systems; (4) the potential for cross-zone fire propagation; (5) the effectiveness of fire barriers; (6) the effectiveness of automatic and manual fire suppression methods; (7) component fragility to fire and combustion products; and (8) hazards associated with transient combustibles.

The results of these licensee assessments have varied widely. Fire events have been found to contribute, typically, from 3 percent to 80 percent of the overall frequency of core damage resulting from internal and external events. For the submittals reviewed thus far, the reported core damage frequencies (CDFs) due to fires range from 1.0E-09 per reactor year to 5.2E-03 per reactor year, with the majority in the range of 1.0E-06 to 1.0E-05 per reactor year. This broad range of reported CDFs is attributed primarily to the methodology and assumptions used for the analyses--such as assumptions related to the treatment of fire barriers, automatic and manual fire suppression, transient fire hazards, and the selection of heat release rates for plant equipment. In addition, considerable variance arises from the safe shutdown methods specific to each facility, including the types of equipment present at the remote shutdown panels and the number and complexity of related operator actions.

For those licensees who committed the necessary resources to perform thorough and intrusive fire risk assessments, these analyses have provided a considerable amount of useful information on the risk posed by fire to their facilities. The assessments have shown certain areas of the plant, such as the control room, the cable spreading room, and the essential switchgear or relay rooms, to be universally risk significant. In most scenarios analyzed, for a fire to lead to core damage, a random failure of equipment must occur, unrelated to the effects of the fire. This finding helps to confirm the essential usefulness of the existing NRC fire protection requirements. In fact, these assessments have substantiated the fundamental success of NRC fire protection requirements in reducing risk to the public and the environment from internal fire events at nuclear power plants--by as much as an order of magnitude. In addition, based on the insights gained from these analyses, some licensees have been able to implement, voluntarily, low-cost plant modifications or procedural changes to reduce their facility-specific CDF for the risk-significant fire areas.

However, the overall variability in the methods and assumptions used for the fire risk assessments, coupled with the uncertainty associated with some of the assessment tools used (such as fire modeling and human factors), have precluded the use of these analyses for a comparison of the fire risk among plants. In addition, this variability in the analyses has limited the application of these assessments for regulatory activities, such as the inspection of licensee facilities, the justification for deviations from NRC fire protection requirements, and--at least to date, although this is being discussed--the development of a risk-informed, performance-based fire protection regulation. Benefiting fully from fire-protection-related risk information will require a reduction in uncertainties related to method, improved gathering of event-related data, and the performance and incorporation of additional research.

In at least a few cases, these IPEEEs, as they are called, <u>have</u> revealed more significant vulnerabilities related to fire protection. I will discuss briefly a specific example. The IPEEE performed at the Quad Cities Nuclear Power Station (an plant of older BWR design near Chicago, Illinois) identified a CDF due to fire events of 5E-03 per reactor year. As a 2-unit facility, the Quad Cities licensee placed a high degree of fire safety reliance on the availability of plant equipment from the opposite unit to allow safe shutdown of the fire-affected unit. The primary factor for the high CDF was attributed to the postulated outage-induced unavailability of safe shutdown equipment, and the large number of manual operator actions needed to achieve and to maintain safe shutdown conditions after a fire. The licensee implemented an interim alternate shutdown method at both units, which reduced the fire-event-related CDF from 5E-03 per reactor year to 9E-04 per reactor year. However, a series of NRC inspections, follow-up discussions, and licensee internal reviews continued to identify new issues. In September 1997, based on deficiencies in the Quad Cities post-fire safe shutdown procedures and a lack of confidence in the associated circuit analysis, the licensee declared all Appendix R post-fire safe shutdown paths inoperable, and voluntarily shut down Unit 2. In December 1997, the licensee also voluntarily shut down Unit 1, based on the lack of NRC confidence in (1) the improved safe shutdown methodology implemented by the licensee, and (2) the operability assessment supporting continued Unit 1 operation.

On January 16, 1997, the NRC issued to the licensee a “confirmatory action letter” (CAL)--a document which emphasizes and confirms, after discussions with the licensee, the NRC understanding of specific actions which the licensee has committed to take in response to an emergent problem. This Quad Cities CAL confirms, among other actions, that the licensee (1) will revise, review, and validate the Quad Cities Safe Shutdown Analysis (SSA) to ensure that a fire will not prevent the performance of necessary post-fire safe shutdown functions; (2) will conform facility safe shutdown procedures to the revised SSA; and (3) will ensure that appropriate plant personnel are trained in using these procedures.

IV. Current Commission Considerations Regarding the Regulation of Fire Protection

I now would like to discuss current Commission considerations related to the regulation of fire protection. In recent years, a great deal of discussion and thought has been given to a wholesale revision of the NRC Fire Protection framework--a revision that would eliminate the plant-specific exemptions to current regulations, incorporate the results of insights from risk assessments, and, perhaps, take a less prescriptive, more performance-based regulatory approach. In 1995, the Nuclear Energy Institute (NEI), an organization that represents the U.S. nuclear power industry, petitioned the NRC to request an amendment to the fire protection regulations. NEI proposed the addition of a new appendix to the NRC regulations that would provide more licensee flexibility while meeting the overall safety objectives of Appendix R, incorporating the insights from risk assessments, and providing a more effective focus for NRC and licensee resources. Ultimately, however, the Commission decided that the NEI proposal did not meet Commission expectations for a risk-informed, performance-based regulation.

In June of last year, the NRC staff submitted to the Commission a paper that described the actions needed to develop a rulemaking that would use a more risk-informed, performance-based structure for fire protection regulation of nuclear power plants. The proposed benefits of such a rulemaking would be to evaluate the safety impact of proposed plant changes in an integrated manner, to reduce the regulatory burden on licensees where requirements were unnecessarily restrictive, and to identify any areas where requirements should be increased. The paper also stated that a research plan would be developed to advance the state of the art in fire modeling and fire risk assessment methods.

In September 1997, the Commission directed the staff to prepare an expedited schedule for this rulemaking, to complete the research needed to support this rulemaking by the end of the year, and to accelerate the resolution of issues, as necessary, to ensure that the proposed rule would eliminate the need for most Appendix R exemptions. The Commission also asked the staff to obtain feedback from the U.S. nuclear power industry on their position with respect to a new rule.

Just last month, the NRC staff presented the follow-up paper to the Commission and provided an associated Commission briefing. In general, the staff found that the U.S. nuclear power industry discounts the need for a new rule, and specifically opposes any rulemaking that would invalidate existing fire protection exemptions. The industry supports NRC clarification of the existing rule and associated documents, more NRC guidance on fire protection engineering evaluations, and more allowance for risk significance evaluations within the current rule.

As an additional consideration, the National Fire Protection Association (NFPA) has advised the Commission that they are developing a performance-based standard for the fire protection of light-water reactors. In a briefing before the NRC Advisory Committee on Reactor Safeguards (ACRS), the NFPA presented a plan in which this fire protection standard would be completed by May 2000, and stated their belief that the NRC should both help to prepare and subsequently adopt this forthcoming standard. In recent months, the ACRS has also been briefed by two public interest groups--the Union of Concerned Scientists and the Nuclear Information and Resource Service--as well as by the Nuclear Energy Institute, regarding the opinion of each organization on the NRC regulatory framework for the fire protection.

If this sounds like a complicated decision-making process, let me assure you--it is. It is characteristic of Commission actions that we endeavor to include, to the maximum extent practicable, the opinions and input of all our stakeholders--including the Congress, other government agencies, professional groups, the licensees we regulate, and the public. The Congress has asked government agencies to work with the industries they regulate, where feasible, in the development of generic codes and standards. We believe fundamentally that, while this process may, at times, be cumbersome, it ultimately will result in a well-informed, responsible, and technically sound regulatory framework for all concerned.

The regulation of fire protection is no exception. In the coming weeks, the Commission will deliberate over several possible options for the future: (1) to develop, on an expedited schedule, a performance-based, risk-informed fire protection regulation to replace the existing regulation, with detailed guidance, and alternatives that would constitute compliance; (2) to defer rulemaking at this time, working instead with the NFPA and the nuclear power industry to develop a performance-based, risk-informed standard for fire protection at nuclear power plants--which may subsequently be endorsed by the NRC; (3) to maintain and enhance existing fire protection regulations and guidance; or (4) some variation on one of these options. All of these options currently are under active consideration by the Commission.

V. Conclusion

In summary, the regulation of fire protection in the U.S. nuclear power industry continues to be an evolving and dynamic process. Several elements of the process, however, are clear. First, the history of fire safety and fire-related events at nuclear power plants, both in the U.S. and in other countries, has provided a wealth of insight into both the safety implications of, and preventive and mitigative measures for, fires at nuclear power facilities. Second, I believe that, in the U.S., we have achieved some measure of success in establishing a regulatory framework that has reduced significantly the risks associated with these events. And third, in considering additional refinements to this framework, the Commission will endeavor to pursue the most effective, efficient, and risk-informed direction possible, as a means to ensure that the protection of public health and safety remains paramount.

I hope that this presentation has enhanced your overall appreciation of the history of U.S. fire protection in nuclear power plants, the associated regulatory framework, and the current deliberations for possible improvements. Even more, I hope that this brief history and description of current activity have reaffirmed, for this audience, the value of international cooperation and information exchange--not only on fire protection issues, but on all matters related to nuclear safety.

Thank you.