Forging a New Nuclear Safety Construct

The ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events

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Prepared by:

The ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events

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FOREWORD

On March 11, 2011, the Great East Japan Earthquake unleashed seismic shocks and tsunami waves of unprecedented magnitudes from the Pacific Ocean, inflicting devastating damage to the nation of Japan. Key infrastructure was destroyed along nearly 650 kilometers or 400 miles of coastline, with major impacts to several nuclear power plants, thermal power plants, dams, oil refineries, the electric power grid, trains, highways, airports, shipyards, manufacturing facilities, and to entire townships. More than 125,000 building structures were ruined and over 300,000 people were left homeless. Most unfortunately, as of March 12, 2012, there have been 15,854 confirmed deaths, 26,992 injured, and 3,155 people missing across twenty Prefectures from the widespread devastation inflicted by the earthquakes and the multiple tsunamis.

We have been and continue to be deeply saddened by the great losses and continued suffering of the Japanese people. More than 650 Japanese engineers participate as members of ASME, and more than 50 Japanese engineers are actively engaged in the development of ASME Standards and Certification programs. Therefore, we have been in direct contact with our colleagues in the Japan Society of Mechanical Engineers (JSME) since the day of the event, offering our help and support. While a team of ASME leaders has been engaged with JSME colleagues in working on the potential impact of the Japan events on nuclear codes and standards since spring 2011, a greater need exists to identify broader lessons learned, particularly from the impact of the earthquake and tsunami on the status and future of all Japan nuclear power plants, as well as its potential impact on worldwide energy portfolios and nuclear power deployment.

It was exactly 100 years ago that members of the newly formed ASME Boiler Code Committee began to meet in New York to address major public outcry for someone to eradicate deadly boiler explosions that were occurring frequently in factories, schools, churches and other locations, with serious consequential human loss, suffering, and economic impact. George Westinghouse had just finished serving as President of ASME in 1910-1911, and the ASME had gained significant respect and stature from its first 30 years of work, particularly with Thomas Edison, Henry Ford, and other industrialists actively engaged as ASME members. The ASME Boiler Code Committee worked hard at reaching a consensus from engineers and other jurisdictions in the United States (U.S.), prior to issuing the first ASME Boiler Code in December 1914. This document provided comprehensive requirements for boiler design, construction, and operation. Shortly after its publication, the ASME Boiler Code was used by several states and local jurisdictions. As a result, the number of boiler explosions plummeted, and this effort was recognized as one of the top 10 engineering achievements of the 20th century. Undoubtedly, the Boiler Code was a successful beginning to an era of technological achievements that require standards and codes for their useful implementation by society.

Today, once again, we find ourselves facing major events and accidents—but under different circumstances. The recent events in Japan come on the heels of other recent catastrophes, including the Deepwater Horizon accident, the San Bruno (California) gas pipeline explosion, mining accidents, and a major dam failure in Russia. With more than seven billion people now inhabiting the globe with increasing needs for energy to sustain or improve their quality of life, we find technological advances pushing limits on many fronts—we are digging and drilling deeper, facilities are operating at higher temperatures and pressures and for longer periods of service, and technology has become vastly more complex and highly interconnected. In fact, ASME recently published a report titled, "Initiative to Address Complex System Failure: Prevention and Mitigation of Consequences," June 2011, initiated after the Deepwater Horizon accident.

The events at the Japan Fukushima Dai-ichi reactors were occurring as the ASME Complex System Failure work was nearing completion. To this end, I appointed an ASME Presidential Task Force to

- review events that occurred and subsequent activities undertaken in Japan and the U.S.,
- develop and disseminate its perspective on the impact of these events on the future direction of the nuclear power industry, and
- make recommendations on ASME's role in addressing issues and lessons learned from these events.

I was pleased that Dr. Nils Diaz, Past Chairman of the U.S. Nuclear Regulatory Commission, and Dr. Regis Matzie, former Senior Vice President and Chief Technology Officer at Westinghouse Electric Company, agreed to serve as Chair and Vice Chair, respectively, of the ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events. This report represents the collective opinion of the group of experts brought together to form the Task Force and evaluate this challenging topic. On behalf of ASME, I want to thank the members of the Task Force for their service and dedication to this vital study. The Task Force review and recommendations provided in this report will hopefully launch activities within ASME, and working with other professional engineering societies, industry organizations, and government agencies worldwide, recommend global actions to prevent and mitigate the consequences of severe nuclear accidents, in a manner similar to ASME efforts a century ago.

Victoria A. Rockwell

Victoria A. Rockwell

ASME President 2011-2012

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EXECUTIVE SUMMARY

The March 2011 Great East Japan Earthquake and Tsunami caused great loss of life and property in the Nation of Japan, and devastation to the environment. The extraordinary forces and flooding unleashed on the East coastal area also led to severe nuclear plant damage and radiological releases at the Fukushima Dai-ichi station. The global impact of the accident at Fukushima prompted the ASME President, Victoria Rockwell, to commission a Presidential Task Force to examine those nuclear plant events and their implications.

The unprecedented accident at Fukushima exposed new information on nuclear power plant vulnerabilities to extreme external events and exposed the need for pertinent improvements. The multi-unit nuclear plant accident at Fukushima continues to have serious impacts on socio-political, economic, and energy-related issues in Japan and worldwide. Within its broad charter, the ASME Task Force chose to build on the growing body of U.S. and international technical assessments of these events, and to examine the Fukushima Dai-ichi accident in the context of the broader lessons learned from a half-century of nuclear operations. From the combination of assessments and reviews of the critical elements involved in the accident scenarios, the ASME Task Force proposes a cohesive framework for continued safe operation of nuclear plants.

Fukushima Dai-ichi – In Context

From the vast body of observations, analyses, and reporting of the events at Fukushima, several points stand out as the most salient factors in assessing its long-term implications:

- The Fukushima Dai-ichi units are the first nuclear reactors in the world in the fifty-plus years of nuclear plant operation to sustain core degradation due to catastrophic external events, the first to involve simultaneous multiple unit failures, and the first light water reactors to release large amounts of radioactivity to the environment.
- The reasons why four of the Fukushima Dai-ichi units were severely damaged, and three suffered core meltdowns, and why ten other nuclear plants in the affected areas were able to survive, are clear and correctible. Principal among those were the now recognized inadequacies in plant design basis for tsunamis, flooding, and accident management.
- The Fukushima Dai-ichi accident reveals no fatal flaw in nuclear technology, yet multiple important safety improvements are being addressed by the global nuclear fleet from the new lessons learned.
- Extensive evaluations of the Fukushima Dai-ichi accident confirm the absence of prompt fatalities from radiological effects and the continuing expectation of no delayed radiological public health effects. The relatively low potential for radiological health consequences from the Fukushima accident is consistent with actual experience with radiation effects.
- Protection of public health and safety from radiological releases has been and continues to be the primary focus of reactor safety. However, past and present experience shows that **the major consequences of severe accidents at nuclear power plants have been socio-political and economic disruptions inflicting enormous cost to society**. As of this writing, 15 months after the March 2011 disaster, about 90,000 residents in that region are still not able to return to their homes, pending a more complete radiological cleanup. As of May 5, 2012, all of the 54 nuclear power plants in Japan are shut down, and the nation continues to struggle with an energy supply shortfall. Estimates of the overall economic consequences of the Fukushima Dai-ichi accident are on the order of half a trillion U.S. dollars.

*Socio-political and economic consequences such as experienced in Japan after the Fukushima accident, even if caused by extreme natural disaster, are **unacceptable.***

After a detailed review, such consequences—even though caused by an extreme and, in some respects, unprecedented natural disaster—appear preventable and are unacceptable, and they are wholly inconsistent with an economically-viable and socially-acceptable use of nuclear energy.

The primary nuclear power safety goal is and will continue to be protection of public health and safety. However, the Fukushima Dai-ichi accident revealed the need for additional steps to further reduce the potential for socio-political and economic consequences resulting from radioactivity releases. On that basis, the ASME Task Force has proposed a new nuclear safety construct to effect such improvement.

A New Nuclear Safety Construct

The set of planned, coordinated, and implemented systems ensuring that nuclear plants are designed, constructed, operated, and managed to prevent extensive societal disruption caused by radioactive releases from accidents, using an all-risk approach.

Critical Elements of the New Safety Construct

- **It is founded on the existing nuclear safety construct**. The new construct will expand on the evolving safety frameworks, reaching beyond adequate protection of public health and safety to prevent socio-political and economic consequences from a severe nuclear accident.
- **It extends the design basis to consider all risks, and includes rare yet credible events**. The ASME Task Force proposes that the new safety construct be based on an "all-risk" approach, addressing a broad range of challenges to nuclear power plant safety, including internal and external hazards, during all modes of plant operation, evaluated in a risk-informed manner. "Cliff-edge" events–those for which a small incremental increase in severity can yield a disproportionate increase in consequences–should be discovered and mitigation approaches implemented. The objective in addressing rare events with potentially extreme consequences is to take reasonable and practical measures to deal with credible events that until now have not been fully considered, while realizing that the overall risk will not be zero.
- **It extends beyond regulations**. It is the ASME Task Force view that accountability for protection of people and property must extend beyond the regulatory requirements to plant designers, manufacturers, owners and operators.
- **It must be embraced globally**. The ASME Task Force recognizes the inherent difficulty in applying any standard across different corporate and regulatory regimes and cultures, but the reality remains—as evidenced starkly by the Fukushima Dai-ichi accident—that the viability of nuclear energy is a global proposition, and that safety principles apply to all plants.

*Safety principles are universally applicable—a **global** safety construct is needed.*

The Challenges

The ambitious objective set by the ASME Task Force—to develop, adopt and support the implementation of a new nuclear safety construct—presents daunting challenges. Principal among these is **building global consensus** on its principles, details, and implementation.

Among the many owners, operators and regulators of nuclear plants around the world there are differences in culture, regulatory structure, technical sophistication, government involvement, economy, environment, politics, and the like. The ASME Task Force recognizes these differences and their implications on the practical work of developing a consensus on the level and nature of extreme events against which the plant, people and property must be protected. Nevertheless, it will be necessary to work together, to find the common denominators, and to achieve global alignment on the fundamentals of nuclear safety.

*Nuclear energy can be economically viable **provided** Fukushima-like consequences are prevented in the future.*

This initiative could be perceived as just another layer of requirements limiting the economic viability of nuclear power. On the contrary, the intention is to support the overall viability of safe nuclear generation. The ASME Task Force is convinced that a new nuclear safety construct can be developed that addresses the safety issues from the Fukushima lessons learned with reasonable and well-defined provisions. Unless consequences such as those experienced in Japan can be avoided, even when confronted with extreme natural events, nuclear power will not be socially-acceptable and economically-viable over the long term. **Building public trust** is an essential component in prevention of adverse socio-political and economic consequences from nuclear plant accidents.

Next Steps

From this starting point begins the real work of building consensus, developing the New Nuclear Safety Construct in full detail, determining the roles of the various stakeholders, and then adopting and implementing it globally. The ASME Task Force report recommends a set of next actions in this regard, particularly using the experience, stature and capabilities of ASME in convening workshops, to bring together worldwide stakeholders including industry, regulators, professional societies, government agencies and industry organizations worldwide.

1 BACKGROUND, SCOPE, AND NEW DIRECTION

1.1 Introduction

The Great East Japan Earthquake and Tsunami inflicted major loss of life and destruction of property on the Nation of Japan, as well as substantial devastation of its environment. Emergency response capabilities were overtaxed and often overwhelmed. Furthermore, the extraordinary natural forces unleashed on the East Japan coastal areas led to a series of accident-initiating events that resulted in the inability to cool the reactor cores in three operating units of the Fukushima Dai-ichi Nuclear Plant, also referred to as Fukushima. Loss of core cooling in Units 1, 2, and 3 led to core degradation and fuel melting. Subsequently, continuing lack of core cooling led to loss of the reactor coolant pressure boundary, loss of containment integrity, and hydrogen explosions from zirconium cladding-water reactions, followed by large radioactivity releases to the environment from all three units. Radiological protection of the public necessitated evacuation of populated areas up to 30 km or nearly 19 miles from the plant.

Due to the current and expected absence of discernible radiation health effects, radiological protection of public health and safety appears to have been effective in Japan; however, the multi-unit nuclear-plant accident at Fukushima continues to have serious impacts on socio-political, economic and energy-related issues in Japan, as well as globally, and has received extensive Government and media attention worldwide. The accident at the Fukushima plant has already affected energy portfolios by skewing the importance of nuclear electricity generation and its beneficial impacts on fuel diversification, climate change initiatives, and stability of electrical costs.

The ASME Presidential Task Force on Response to Japan Nuclear Power Events (ASME Task Force) is convinced that global and thoughtful solutions to the issues raised by the Fukushima Dai-ichi nuclear accident are essential to continue benefitting from use of nuclear power, to expand its use, and to address critical environmental and energy portfolio issues. The ASME Task Force is proposing an extended safety framework that would add complementary improvements to the existing nuclear safety infrastructure in a systematic manner, strengthening safety and accident response to external and internal events. The proposed improvements are focused on prevention or minimization of major impacts on public health, the environment, and socio-political-economic issues from large accidental releases of radioactivity. To achieve these objectives, the ASME Task Force supports development of a new safety construct for nuclear power generation, as described in this report, which enhances the existing safety framework using an all-risk approach.

The term “construct” denotes the conjunction of component parts that, working together, achieve a desired outcome. For nuclear power plants, the existing and evolving safety construct incorporates a set of elements, including plant design, physical systems, structures, and components, safety regulations, quality assurance, and procedures and practices for plant operation and maintenance, accident management, and emergency preparedness.

The term “all-risk,” as used in this report, refers to consideration of all credible hazards in developing probabilistic risk assessments (PRAs), assessing defense-in-depth, and developing accident management strategies. Risk is the combination of the probability of an adverse event and its consequences. Protection of public health and safety and the environment are and will continue to be the most important consideration for nuclear safety. Complements to the existing and evolving safety construct would further strengthen the protection of public health and safety. The additional consequence of concern in the new safety construct is extensive disruption of society from a radioactivity release to the environment.

"All risks" should be considered to include rare yet credible events and potential accident scenarios that could threaten the safety of a nuclear power plant. Accident scenarios can be initiated by either internal or external hazards from natural or man-made causes, during all modes of plant operation.

Addressing rare yet credible events with extreme consequences should be limited by appropriate risk considerations. In this regard, it is important to rely on the rule of law, using what the Courts have determined is acceptable for nuclear power safety: *"The level of adequate protection need not, and almost certainly will not, be the level of zero risk."* [1]

Events that are not credible need not be considered. It is the intention of the ASME Task Force that very-low-probability events, such as extreme floods, large scale tornadoes, and other natural phenomena that are unprecedented but conceivable at a given site, should be considered. Of particular concern are initiating events that could lead to cliff-edge effects, whereby for a small incremental increase in severity, the consequences disproportionally increase. In those cases, systems and planned actions should be in place to provide core cooling and prevent a large release of radioactivity.

1.2 ASME Task Force on Response to Japan Nuclear Power Events

ASME President Victoria Rockwell established a Presidential Task Force to identify the Society's role in addressing the multi-disciplinary issues posed by the aftermath of the Fukushima accident, and to recommend global actions to prevent and mitigate the consequences of severe nuclear accidents. The ASME Task Force has not conducted its own detailed assessment of the Fukushima Dai-ichi accident's progression. Instead, the ASME Task Force has relied on the well-scrutinized assessments conducted by others, including Japan's Government authorities, industry, and qualified academic and professional institutions, the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD), various regulatory authorities (including the U.S. Nuclear Regulatory Commission (NRC), and the European Community regulatory agencies), the Institute of Nuclear Power Operations (INPO), the Electrical Power Research Institute (EPRI), the Nuclear Energy Institute (NEI), and the American Nuclear Society (ANS).

The ASME Task Force focused first on the technical elements contributing to the accident and its consequences, and then concentrated its efforts on possible solutions to the emerging lessons learned from Fukushima, while considering other major nuclear power accidents and incidents. A new approach, which considered the need to avoid significant socio-political and economic consequences, in addition to protection of public health and safety, emerged as the principal theme of the report.

1.3 The Fukushima Dai-ichi Nuclear Accident

The Fukushima Dai-ichi units are the first nuclear reactors in the world to experience core degradation due to a catastrophic external event, and the first light water reactors (LWRs) to experience accidents resulting in large radioactivity releases to the environment. Furthermore, the accident at Fukushima was the first multi-unit accident in power reactor history, complicating response and recovery. Because containment integrity was maintained, the Three Mile Island Unit 2 (TMI-2) light water reactor accident, that experienced a partial core meltdown, resulted in no significant releases of radioactivity to the environment. On the other hand, the Chernobyl disaster, which involved a reactivity excursion in a graphite-moderated, water-cooled reactor, resulted in a steam explosion, burning of the hot graphite in the core, and, absent a containment structure, release of the radioactive contents of the uranium fuel to the environment. The TMI-2, Chernobyl, and Fukushima Dai-ichi reactor accidents have common features that are explored in this report. These experiences show that maintenance of core cooling—before or during progression of an accident—must take priority before other accident management activities.

The external initiating events (combined earthquake and tsunami) of the Fukushima Dai-ichi accident were significantly beyond the plants' design capabilities, resulting in a complete loss of alternating current (AC) and direct current (DC) power, and loss of ultimate heat sink. [2] The reactor operators were eventually left with no readily-available means to continue cooling the reactor cores, and therefore were unable to fulfill the most important of all safety functions for nuclear power plants. The ensuing combination of severe damage to the core and the containment at three of the six Fukushima Dai-chi units led to loss of containment integrity and to significant radiological releases. Those releases could have been avoided had the Fukushima Dai-ichi Plant been better protected from severe external events, and supported by on-site and off-site resources to restore cooling prior to melting of the fuel in the reactor cores and loss of the integrity of the containments.

1.4 The Accident's Outcome

The public health outcome of the Fukushima Dai-ichi nuclear accident, from a radiological protection perspective, resulted in no prompt fatalities and the continuing expectation of no significant delayed radiological public health effects. [3] However, there were significant consequences, including radiological contamination of a large populated area in Japan, initial relocation of more than 100,000 people for radiological protection purposes, extended loss of economic productivity of the contaminated areas, wholesale curtailing of nuclear power generation across Japan, and accompanying economic impact. Furthermore, the worldwide reaction to the consequences of the accident has serious economic and energy strategy implications, including consideration of the role of nuclear power for future energy supply.

Reliable estimates of the economic cost of the Fukushima accident are difficult to ascertain, although easier to quantify than the social impacts and the political ramifications, for Japan and globally. The current rough estimate of the total cost to Japan from the Fukushima Dai-ichi accident is about $500 billion U.S. dollars, which will substantially increase if nuclear electricity generation continues to be replaced for a long time by other means. Appendix A includes summaries of the estimated impacts of the accidents at Fukushima, Chernobyl, and Three Mile Island.

The economic cost for the Fukushima Dai-ichi accident is commensurate with the wide estimates given for the Chernobyl accident, which range from $250 billion to $500 billion U.S. dollars over the last 25 years. [4] The socio-political consequences of the Chernobyl accident have been found to dominate the effects on the population and the countries affected.

The direct economic impacts of the TMI-2 accident have been estimated to be several billion U.S. dollars; however, the socio-political impacts have not been ascertained. Probably the best known impact from TMI-2 is on the electricity portfolio in the U.S.; i.e., TMI-2 contributed, along with financial and other factors, to nuclear power plant construction cancellations and limited growth of nuclear generation, for over 30 years. Nuclear deployment is only now beginning again in the U.S., with near-term construction projects in Georgia and South Carolina.

Day-to-day operation of nuclear power plants provides significant benefits to society with lower overall risk to public health and safety than the majority of other energy sources. Nuclear power operations are environmentally benign and produce large amounts of electricity at a low production cost. Nuclear generation has a lower overall lifetime health risk than many other complex technologies, even when the potential for severe accidents such as Fukushima Dai-ichi are taken into account. However, large radiological releases, like the ones from Chernobyl and Fukushima, are different from the usually-more-localized risk from non-nuclear industrial complexes. The real and perceived consequences from a severe reactor accident with significant offsite releases of radioactivity are distributed and long-term, and could impact large geographic areas with a variety of effects, often driven by the fear of small radiation exposures out of proportion to the actual risk.

Furthermore, the media coverage and public reactions are also vastly different from what would be expected from non-nuclear industrial risks and further exacerbate the impact on society.

The widespread consequences of the accident at Fukushima are central to the ASME Task Force's proposal for actions to ensure continued safe and reliable operation of existing and new nuclear power plants. Protection of public health and safety from radiological releases has been and continues to be the primary focus of nuclear safety. The present body of knowledge, including lessons from severe reactor accidents, establishes the importance of maintaining that focus while bringing about another important realization: **The major consequences of severe accidents at nuclear plants have been socio-political and economic disruptions inflicting enormous cost to society.** In other words, even when there are no discernible radiological public health effects from a nuclear power accident, the observed and potential disruption of the socio-economic fabric of society from a large release of radioactivity is not an acceptable outcome. Therefore, there is a compelling reason to develop a new safety construct for the nuclear power industry, explained in Section 1.6 below.

1.5 Key Issues and Scope of Work

The ASME Task Force begins this report with a brief historical perspective of major reactor accidents with radiological consequences and other events with significant safety importance, as well as the safety improvements they engendered. The key issues that commonly appear in that history are present before, during and after the Fukushima Dai-ichi accident. They include: core cooling, reactor coolant pressure boundary integrity, containment integrity, containment and capture of fission products, reactivity control, human performance, safety instrumentation and control, communications, command and control, and emergency preparedness.

A salient issue arising from the accident at the Fukushima Dai-ichi Plant is the need for adequate protection from catastrophic external events that exceed the design basis requirements. Therefore, severe-accident management should include events previously termed "beyond-design-basis." This term implies a state of severe challenge to safety systems that could result in a large uncontrolled release of radioactivity from rare yet credible initiating events, also called low-probability, high-consequence events. In this report, severe accident management and the established design basis are first reviewed within the context of the powerful statutory U.S. requirement for "reasonable assurance of adequate protection." The capability to cope with threats to or complete failure of safety systems is addressed from the perspectives of accident prevention, transition to failure (accident interdiction), and accident mitigation. The importance of defense-in-depth and risk-informed insights, and the interaction between these complementary methodologies, is incorporated in the discussion, with a primary focus on core cooling. The additional capabilities under consideration to extend the design basis to cover severe accident response are then discussed, and the advantages of combining probabilistic risk assessment with defense-in-depth are described.

From the standpoint of safety system performance, the lessons from the Fukushima Dai-ichi accident are patent and relevant, even if the time-dependent behavior of the reactor coolant system and the safety systems during the early phases of the accident are not completely known. The urgent lessons learned require establishing or improving the safety performance of systems, structures and components to severe external events that threaten the performance of safety functions such as coping with station blackout, under all conditions, and maintaining the availability of the ultimate heat sink. As is the case in all countries with nuclear power plants, the NRC has been conducting exhaustive reviews of the accident, establishing lessons learned, and proposing recommendations for safety improvements at U.S. nuclear power plants. [5] The U.S. has already established complementary capabilities to perform safety functions after "large fires and explosions" under the Interim Compensatory Measures issued as the "B.5.b requirements" and codified at Title 10 CFR Part 50.54 (hh), after the September 11, 2001 terrorist attacks. These measures are being used in defense of severe threats to safety systems, and efforts are being directed at their improvement. The interactions among the on-site plant systems, the off-site movable equipment and other accident management capabilities, and the severe accident management infrastructure become extremely important, and they will become more important as an all-risk approach is incorporated into the nuclear safety construct.

The importance of human performance and decision-making is also apparent from the Fukushima Dai-ichi accident. This report discusses the important areas of safety culture and safety management, including their significance for normal operations and accident management. Finally, the report analyzes the evolving role of emergency preparedness in the context of radiological protection, and the importance of public trust on socio-political consequences.

A set of recommendations is centered about the need for the nuclear industry to first ensure that adequate protection of public health and safety is maintained and then to ensure that protections are available to reliably prevent or minimize disastrous societal consequences from an accident's radioactive releases.

1.6 A New Nuclear Safety Construct

The multiple-reactor Fukushima accident has had no significant radiological consequences to the health and safety of the public, due to the protective emergency actions, the inherently slow radioactive releases from the accident, and the fact that the prevailing winds pushed most of the radioactive plumes to the open ocean for over 3 weeks. The combination of these factors contributed to achieving the safety goal of providing radiological protection of the public, in spite of the severity of the accident and the challenges encountered in accident management. [6] However, three reactor cores suffered meltdowns and their containment integrity was lost, with large uncontrolled radioactive releases to the environment. The governing reactor safety criteria of no significant fuel melting and no uncontrolled radioactivity release were not met. Moreover, an extended safety approach is warranted to achieve the requisite overall protection and socio-political acceptance of global nuclear electricity generation. Such an approach, as described in this report, is consistent with practical and achievable improvements to reactor safety and radiological protection of the environment. The approach builds upon the substantial existing safety framework and is focused on achieving support from multiple cognizant organizations on a global basis to forge a new safety construct that would better serve society.

Nuclear power plants have an established safety construct that has been evolving for over 50 years to increase safety and reliability. The ASME Task Force believes the presently-established safety construct constitutes a sound foundation for a more-encompassing safety approach. The emphasis of the following proposed new safety construct is focused on consideration of an all-risk criterion that includes both internal and external threats, whether from natural or man-made events, and further supports protection of public health and safety and the environment.

A New Nuclear Safety Construct

The set of planned, coordinated, and implemented systems ensuring that nuclear plants are designed, constructed, operated, and managed to prevent extensive societal disruption caused by radioactive releases from accidents, using an all-risk approach.

The new safety construct goes beyond the customary regulatory statutes and current best industry practices that are focused on radiological protection of public health and safety and the environment. It establishes a more encompassing safety goal, by inclusion of rare yet credible events into an all-risk approach, and strengthens the emerging safety construct that the global nuclear power industry is developing as a result of the Fukushima Dai-ichi accident.

Essential elements of the new safety construct include:

- Capability to address potential events beyond the design basis and possible cliff-edge effects;
- Confirmation that the design basis or extended design basis includes rare yet credible events;
- Use of an all-risk approach; ensuring core cooling for all phases of accident progression; and
- Improved human performance, organizational infrastructure, command and control, accident management, and emergency preparedness.

Strengthening of prevention and mitigation capabilities is fundamental to the construct. While recognizing the important role that regulatory requirements have and will play in the establishment of the new safety construct, the ASME Task Force believes that the construct would establish demands on the industry that are normally beyond regulatory requirements. Industry implements safety and industry should be in a lead position to effect the changes proposed in the new safety construct. Furthermore, recognizing the different roles of the international nuclear fleet owners and independent regulatory authorities, the ASME Task Force believes that the principles outlined as part of the New Nuclear Safety Construct should be globally applied, with due consideration of the differences between existing and new nuclear power plants.

Figure 1 is a simple characterization of how the New Nuclear Safety Construct builds upon the existing safety construct. While relying on the design basis, it recognizes that additional measures could be needed to address rare yet credible events, and that the best way to accomplish this is to use an all-risk approach. The figure, which is not to scale, shows a progression of safety measures, starting with the design basis; then additional prevention measures from the post-9/11 (September 11, 2001) equipment and accident management improvements; plus the difference in inherent safety features of new reactors; then the safety improvements from the international regulatory and industry post-Fukushima efforts; and, finally, an additional level of coordinated systems to address requirements for avoiding disruption of society. One of the critical issues to be resolved for achieving an effective safety construct is the important role that regulatory authorities will play in resolution of the extended safety basis emerging after Fukushima, and the role of industry in implementing additional safety requirements that satisfy the goal of societal protection. These require coordinated efforts of regulatory authorities, plant owners-operators and other industry stakeholders, with industry taking a leadership role.

The ASME Task Force anticipates that the global nuclear power industry can, in a timely manner, forge a new nuclear safety construct that fits the present body of knowledge and society’s expectations. The ASME Task Force is convinced that a new nuclear safety construct for nuclear power should not be imposed as a regulatory mandate but as the outcome of an in-depth analysis of existing and additional design and accident management capabilities, used and based on the evolving extended regulatory framework. It is the intention of the ASME Task Force that the conclusions and recommendations of this report would serve to promote pertinent actions among global nuclear industry decision-makers and will be supportive of and supported by regulatory authorities.

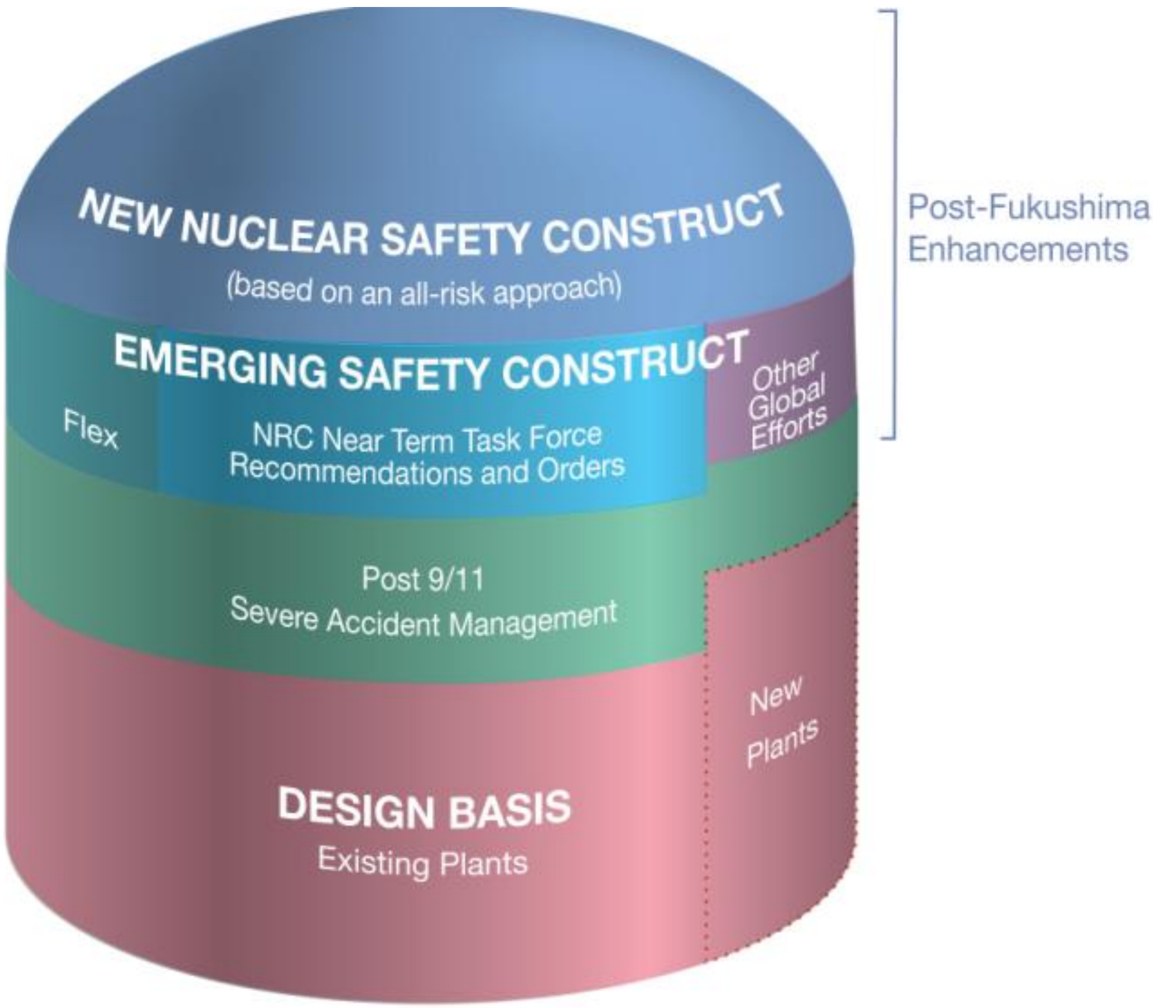

Figure 1 – Forging a New Nuclear Safety Construct

2 AN HISTORICAL PERSPECTIVE ON NUCLEAR SAFETY

2.1 Introduction

Nuclear fission is now a mature technology generating a significant share of the world's ever-growing need for electrical power. As of March 30, 2012, 436 reactors were operating in 31 countries, supplying about 13% of the global electricity consumption. Many countries rely on a much higher share of nuclear-generated electricity for their needs. In most cases, these nuclear power plants operate economically, reliably, cleanly, and safely.

Nevertheless, nuclear power remains a controversial technology. The prevailing view of the 2011 Great East Japan Earthquake and Tsunami—that it was primarily a nuclear disaster despite the enormous toll in human life and habitat destruction completely unrelated to the Fukushima Dai-ichi radiological releases—is a compelling example.

Throughout history, humankind has faced the opportunities and challenges presented by adoption and advancement of technology. The pattern is consistent and predictable—initially-high expectations, followed by early missteps, leading to learning and steady improvement and higher levels of performance and safety. Some technologies exceed expectations, and others fall short. Likewise, some find widespread acceptance and become long-term, stable contributors to society's needs, while others are bypassed or superseded by the next newest technology.

A recent article in The Economist stated "It is not the essential nature of a technology that matters, but its capacity to fit into the social, political, and economic conditions of the day. If a technology fits into the human world in a way that gives it ever more scope for growth, it can succeed beyond the dreams of its pioneers." [7]

Fifty years of ever-expanding applications of nuclear power technology have seen much successful operation, along with multiple incidents and a handful of significant accidents—all leading to growing understanding and lessons learned which in turn have led to improved safety and reliability of operating plants and new plant designs.

The historical perspective presented herein is not intended to suggest a diminished importance or urgency in achieving even higher levels of nuclear plant safety, including prevention and mitigation of accidental releases for protection of public health and property. On the contrary, this report advocates for even stronger steps to prevent large accidental releases of radioactivity, while placing this quest in an appropriate historical context.

2.2 Life Cycle of Complex Technologies

The life cycles of other technologies that have matured to the point that they make major societal contributions and enjoy broad public acceptance—despite significant early challenges—provides context for the present discussion of nuclear power. [8]

2.2.1 A Textbook Example: Boiler and Pressure Vessel Technology [9][10]

Boilers were the revolutionary enablers of industrial progress in the nineteenth century, providing steam power for ships and rail locomotives and serving as prime movers in steel mills, factories, and woodworking shops. The middle to late 1800s saw rapid expansion of boiler applications and escalation in capacity. By 1890, there were some 100,000 commercial boilers in service in the U.S. alone.

Rules and guidelines covering design, manufacture, and operation of steam boilers were non-existent at the time, and failures were commonplace. On April 27, 1865, a boiler explosion on the steamboat

Sultana resulted in one of the worst maritime disasters in U.S. history. More than 1,500 passengers and crew died on that fateful day, roughly the same number as perished on the *Titanic*.

Over the next few decades there were thousands of boiler explosions, many with severe consequences to life and property. It was not until 1911, under the leadership of newly-elected ASME president Colonel E.D. Meier that a viable approach to controlling that unsatisfactory situation was found. Colonel Meier believed that a set of technically-sound requirements formulated by ASME, with its established reputation, independence, and broad scientific interests, could be widely adopted and implemented. At his direction, a small committee of industry volunteers produced the first edition of the ASME Boiler Code, *Rules for the Construction of Stationary Boilers and for Allowable Working Pressure*. That Code later formed the basis for nuclear reactor construction standards used worldwide today.

This publication, issued in 1914, evolved into the ASME Boiler and Pressure Vessel Code (BPVC), which today covers industrial and residential boilers as well as nuclear reactor components (since 1963), transport tanks, and other types of pressure vessels. Through the decades, the BPVC has become virtually synonymous with ASME and has contributed to the organization's stature in the global standards-setting community. The BPVC has been incorporated into the laws of all 50 United States, throughout the provinces of Canada, and in 100 countries around the world.

As the ASME guidance was implemented, the number of boiler explosions steadily declined, even with significant increases in operating pressure (see Figure 2). This work has been recognized as one of the top 10 engineering achievements of the 20th Century. These efforts saved countless lives, gave birth to standards development worldwide, and provided safety benefits to society across a wide range of engineering applications.

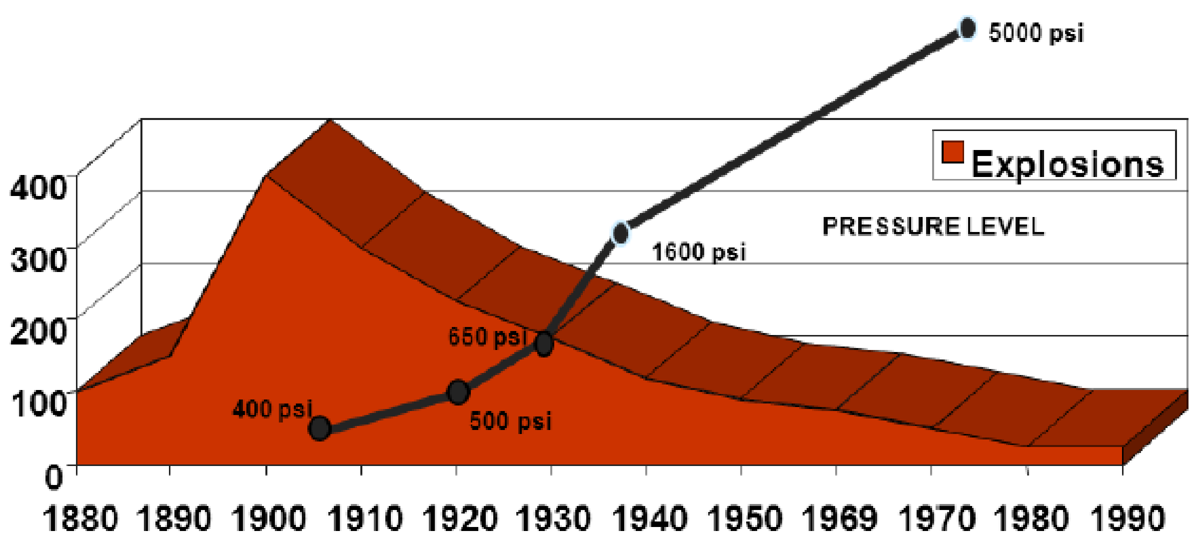

Figure 2 – Boiler Explosion Trends in the U.S.[1]

[1] Based on Statistics Provided by the National Board of Boiler & Pressure Vessel Inspectors

2.2.2 Other Non-Nuclear Examples

There are many other examples of the natural maturation process in virtually all branches of modern technology. Operating experience reveals this consistent pattern in various technologies, including failures in buildings and bridges; in electricity generation, transmission, and distribution; and in air, rail, and automotive transportation.

The path to safe commercial aviation, for example, faced a range of challenges, including application of previously-untapped science and technology, government regulation, wartime military applications that accelerated and stretched the process, and public skepticism. In parallel with rapid learning were continued advancements—in aircraft speed, range, instrumentation, propulsion (propeller to jet), etc. —each introducing more capability and often, additional hazards. As with other technologies, aircraft accidents first increased in number and consequences, but they eventually led to safer aircraft and fewer accidents as the lessons learned were used to improve designs and management systems.

The refinement of the aviation regulatory system—in parallel with ongoing successful development in aircraft technology—yielded the excellent aviation safety record shown in Figure 3 [11]; accidents with fatalities still occur, but at a much lower rate. This performance also reflects implementation of safety initiatives worldwide following the terrorist attacks of September 11, 2001.

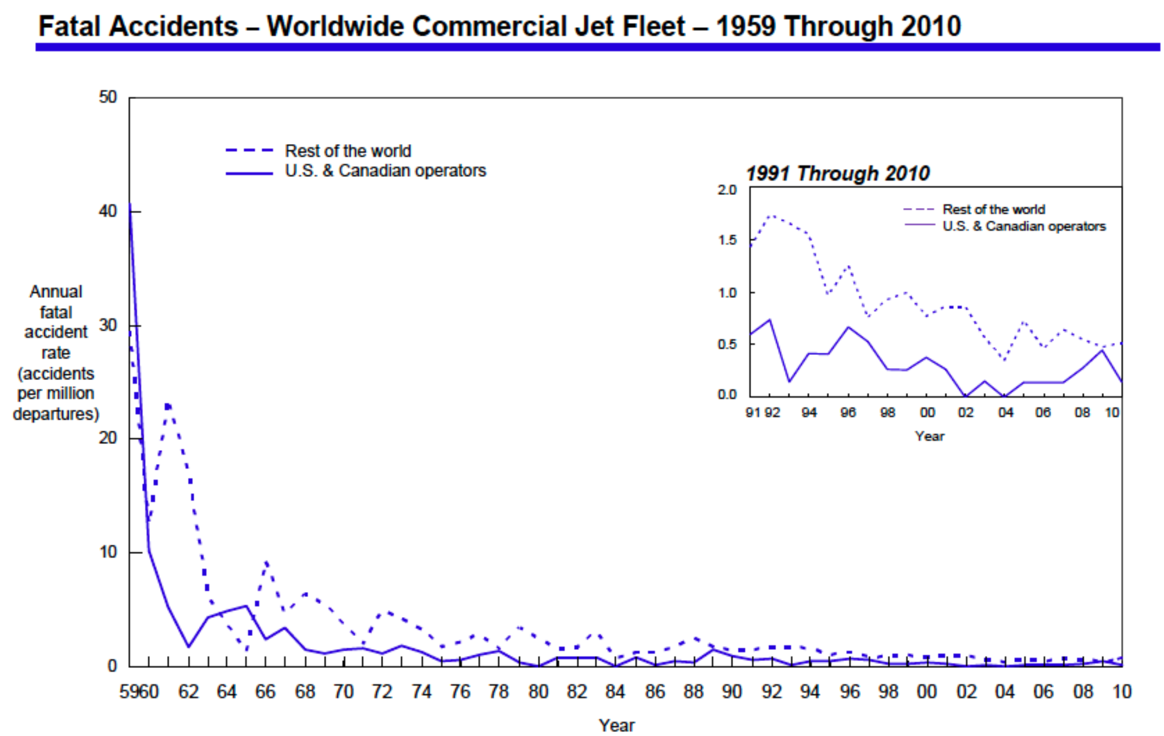

Figure 3 – Commercial Aircraft Accident Rates by Year

Similarly, among the major energy-producing technologies, there is ever-expanding demand for improvement and productivity, and a continuing cycle of learning and refinement based on experience, including serious accidents.

The learning cycle is a continuing cycle; existing and new technologies will have successes and failures that will contribute to their improvements. Recent events include explosion of the Deep

Water Horizon that killed 11 people and caused extensive environmental damage in the Gulf of Mexico and its beaches, and the explosion of the San Bruno natural gas pipeline that took 8 lives and destroyed 38 homes in northern California. In particular, the oil spill in the Gulf of Mexico was a costly accident, with significant socio-economic impact and political ramifications.

It can be expected that the lessons learned from these serious events will result in ever-improved safety in these vital technologies, without cessation of drilling and pipeline deployment.

Challenges of meeting growing energy needs all over the world must be met with improved and economical technologies—there is no alternative. History teaches that with experience and concerted effort, these technologies can and will be harnessed, with improved safety and reliability.

Appendix A includes a summary of historical information on operational performance of nuclear power plants, including data on capacity factors, electricity production costs, and other performance-related factors.

2.3 A Half-Century of Nuclear Experience

As demonstrated by the above examples, experience is an important teacher for both emerging and mature technologies. For nuclear technology, the most profound improvements in its 50-year history have been in operational safety performance—a fact that has led to day-to-day improvements in safety, reliability, efficiency and economy of operating plants worldwide. Of course, there has also been a significant step change in the focus on safety improvements after each major reactor accident, as the lessons learned have been fed back into the design of systems and processes used by the nuclear industry.

To some degree, the industry attention to, and resultant improvement in, plant operations is a consequence of the relative infrequency of serious nuclear safety events in that time frame—itself an indicator of a safe technology, but also a circumstance that inherently limits the learning experience afforded by safety events. Nuclear safety is the prime subject of this report, and learning experiences from nuclear safety events have been significant, as discussed later in this section. But safety and operational performance go hand in hand, and the operational maturation of nuclear plants is significant and reveals much about the strength of nuclear technology.

In the first decade of large-scale commercial operation in the U.S. plant performance was disappointing in many respects. For most plants, installation costs and schedules, plant capacity factors, operating costs, refueling outage times, and in-plant radiation controls were not as favorable as projected. Then, in 1979, the TMI-2 accident—the first major commercial plant safety event—sent shock waves through the nuclear industry.

Following the TMI-2 accident, the owners and operators of all U.S. nuclear plants joined together to address the need for improving operational safety and reliability, and established a self-policing/self-improving organization to achieve excellence in nuclear plant operations. INPO was chartered to examine operations at each U.S. plant, to capture and disseminate lessons learned and best practices, to identify key operational issues, and to establish performance metrics with aggressive new targets of operational excellence. In concert with efforts of other industry organizations, including owners' groups and industry organizations like the NEI and EPRI, these initiatives were embraced by the nuclear plant owners and resulted in a steady improvement in operational and safety performance. Independent parallel efforts by the NRC resulted in a further focus on operational safety, especially effective from the 1990's onward.

Figure 4 below is based on a compilation of data for U.S. nuclear power plants from several sources [12]. These data show a consistently-strong trend of improvement in both operational performance, as measured by fleet capacity factor (with 1 representing 100%), and safety performance, as measured by the number of safety-significant events per plant per year.

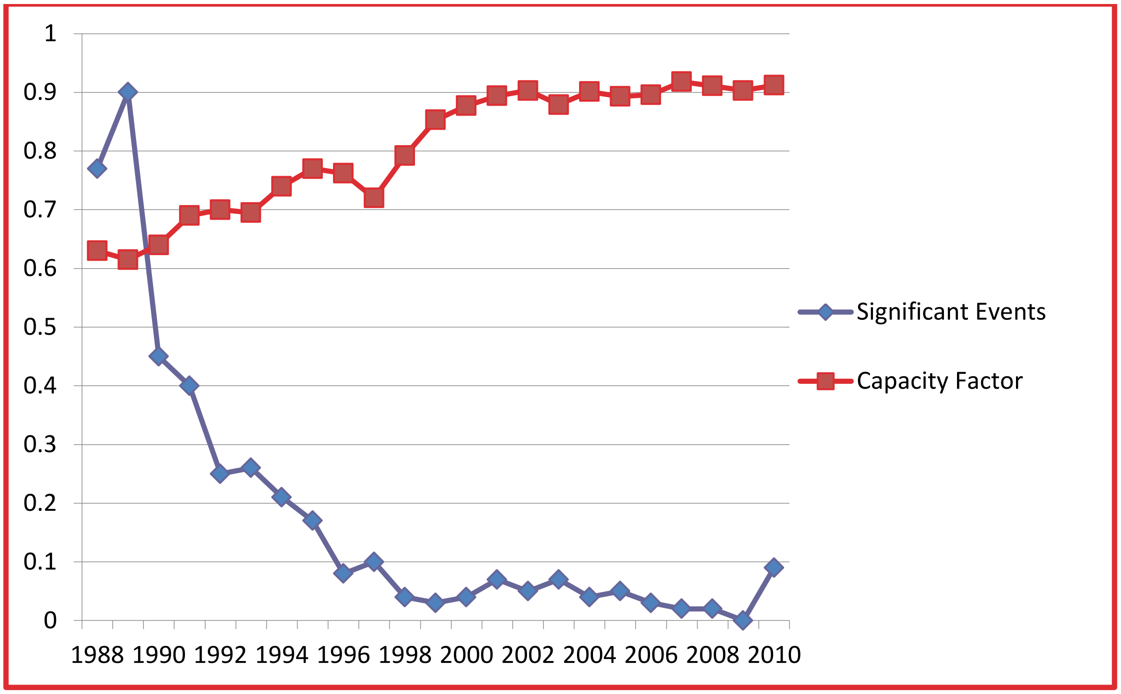

Figure 4 – Safety Significant Events Per Plant and Fleet Capacity Factors

As a prime indicator of the overall success of these industry efforts and the resulting maturation of nuclear power technology, the entire U.S. fleet of 104 operating reactors has achieved a composite 90% capacity factor for the past decade—compared to plant capacity factors routinely in the 60% - 70% range 30 years earlier. In addition, there are no discernible radiological health effects from operations and incidents in U.S. nuclear power plants, including the TMI-2 accident.

2.4 Nuclear Plant Safety

During this increase in operational performance, the industry has produced a significant reduction in safety-significant events, also shown in Figure 4. It is not coincidental that improvements in capacity factor trend with improvements in safety—they both occur as a result of attention to detail, regulatory improvements, and industry sharing of lessons learned and best practices.

In more than six decades of large-scale applications of nuclear technology—including ship propulsion, electrical power generation, nuclear fuel cycle, and weapons production—there have been many incidents but few accidents that led to significant radioactive releases to the environment. While this favorable circumstance has yielded a relative scarcity of accident data to analyze, each incident and accident has provided an opportunity for learning and improvement, making possible ever-safer plant operations

The recent Fukushima Dai-ichi accident, as well as the experience of other nuclear plants in the region able to survive the earthquake and tsunami without core degradation, has opened a new avenue for evaluation of accident management, lessons-learned and corrective actions, such as those described in this report.

The following is a tabulation of major nuclear technology incidents and accidents since the 1950s. In each case, a short summary of the nature of the event is provided, along with its major causes and consequences, and the safety improvements that came from studying its causes. The tabulated events are considered particularly significant from a nuclear safety standpoint, in that they comprise a wide spectrum of causes and consequences and are broadly representative of the 50-year learning process.

Table 1 – Pivotal Events in World Nuclear Experience

Date	Event	Causes and Consequences	Key Outcomes
Oct 1957	**Fire in the Windscale** weapons materials production reactor, UK	Cause: Operation outside established safe regimes and inadequate understanding of physical phenomena, lack of containment. Consequences: Facility destroyed; significant off-site radiological contamination.	Safety culture and reactor containment improvements.
Jan 1961	**Prompt criticality and explosion in U.S. Army SL-1** experimental reactor, USA	Cause: Operator error and inattention; faulty reactor shutdown design. Consequences: Three fatalities within the facility; reactor destroyed; local onsite radiological contamination.	Safety culture and reactor shutdown improvements.
Mar 1975	**Fire in electric cabling of Browns Ferry** Unit 1 power reactor, USA	Cause: Lighted candle used by maintenance engineer to check for firewall air leaks ignited foam sealant in the cable spreading room. Consequences: Single fire knocked out several electrical operating and safety systems; lost reactor power monitoring and emergency core cooling; significant damage to plant controls. No radiological release and no injuries onsite or offsite.	Improvements in construction quality assurance and inspection methods; upgraded fire protection systems, per new Federal regulations (10 CFR 50, Appendix R).
Mar 1979	**Loss of coolant and partial core melt in Three Mile Island** Unit 2 power reactor, USA	Cause: Undetected relief valve failure, compounded by operator error in misinterpreting conditions and terminating emergency cooling, caused loss of coolant, core overheating and partial melting. Consequences: Limited off-site radiological releases (primarily noble gases); no injuries or health consequences; profound loss of public and political confidence; permanent loss of facility; disruption of other nuclear plant licensing and construction and a contributing factor to the cancellation of dozens of new plant orders.	Sweeping changes in training, emergency procedures, human factors engineering, control room design, instrumentation, regulatory oversight and emergency planning. Formation of the Institute of Nuclear Power Operations (INPO) to achieve excellence in U.S. nuclear operation.
Feb 1983	**Two failures of reactor trip breakers at Salem** Unit 1 power reactor, USA	Cause: Equipment design and maintenance shortcomings. Consequences: Reactor tripped manually with no damage.	Industry-wide maintenance improvements and regulation (10 CFR 50.65); safety classification of sub-components; strengthened post-trip review and emergency operating procedures.

Table 1 – Pivotal Events in World Nuclear Experience

Date	Event	Causes and Consequences	Key Outcomes
Apr 1986	**Prompt criticality and explosion, core destruction and fire at Chernobyl Unit 4,** power reactor, Ukraine, USSR	Causes: Core design flaws in RBMK reactors (including positive void coefficient); operator errors in performing test outside plant design basis with safety features disabled; lack of containment; inadequate emergency response. Consequences: Worker fatalities from fire and radiation exposures; facility destroyed and enclosed in a sarcophagus; uncontrolled off-site radiological contamination extending over much of Europe and Belarus: evacuation and resettlement of more than 336,000 people [13] from contaminated areas.	Design changes in RBMKs. IAEA promulgation of "Safety Principles" in INSAG-3 for all nuclear power plant designs, including roles of reactor shutdown design and containment. Establishment of independent nuclear safety regulators in some countries. Formation of the World Association of Nuclear Operators (WANO) to promote consistent standards and excellence in operations worldwide.
Dec 1999	**Flooding at Blayais Station**, power reactors, France	Cause: Water ingress from severe storm and high tides, exceeding seawall height and flooding plant electrical equipment and safety system spaces. Consequences: Loss of offsite power, some safety system power supplies as well as portions of essential cooling water and emergency core cooling systems; very serious near miss event.	Review of design for flooding protection at all EdF stations, extensive upgrades throughout the system.
Sep 2001	**Terrorist attacks** on World Trade Center and Pentagon, USA	Cause: Terrorist attacks on U.S. infrastructure Consequences: Although nuclear facilities were not targeted explicitly, the events of September 11, 2001 sharply increased attention to external factors and events that can impact on nuclear plant safety.	NRC requirements for U.S. nuclear power plants to cope with major external fires, explosions and other external events.
Mar 2002	**Reactor vessel head erosion in Davis Besse** power reactor, USA	Cause: Long term leakage of acidic borated water from crack in control rod drive mechanism; inadequate inspection and maintenance; unsatisfactory safety culture. Consequences: A large cavity nearly penetrating the reactor pressure vessel head, leaving minimal pressure retention capability.	Industry-wide improvements in safety culture; many replacements of reactor vessel heads.
Apr 2003	**Fuel element failures at PAKS Unit 2,** power reactor, Hungary	Cause: Inadequate cooling provided for spent fuel assemblies temporarily placed in a special vessel for de-scaling in the spent fuel pool.	Recognition of inadequate technical knowledge among plant staff and inadequate technical engagement of

Table 1 – Pivotal Events in World Nuclear Experience

Date	Event	Causes and Consequences	Key Outcomes
		Consequence: Fuel overheating, resulting in significant cladding failure, special vessel rupture and fission product release within the plant and modest offsite releases. No discernible offsite impact.	supplier.
Mar 2011	**External event leading to station blackout, loss of core cooling and multiunit core melting at Fukushima Dai-ichi** Units 1-4 power reactors, Japan	Cause: Magnitude 9.0 off-shore earthquake triggered a severe tsunami, flooding the plants and causing full station blackout; inadequate tsunami design basis (including water tightness for essential equipment and inability to readily hookup portable emergency equipment). Consequences: Core melting in 3 units; hydrogen explosions; pressure vessel and primary containment damage; large uncontrolled off-site radiological releases (the largest release of noble gasses in history, and Cs-137 estimated at ¼ to ½ of the release at Chernobyl) [14]; emergency evacuation of approximately 110,000 [15] people; facilities destroyed; extensive cleanup and decommissioning, including cleanup and recovery of contaminated lands; protracted shutdown of nuclear plants in Japan; political ramifications and loss of public confidence worldwide.	Multiple assessments by industry, governments and regulators, worldwide; broad-based industry and regulatory enhancements to safety; emerging and proposed a new nuclear safety construct in this ASME report.
Mar 2011	Great East Japan Earthquake and Tsunami leading to **damage at ten other nuclear power reactors**, Japan	Cause: As above, off-shore earthquake and resultant tsunami caused loss of offsite power and varying degrees of flooding and equipment damage at Fukushima Dai-ichi Units 5&6, Fukushima Dai-ini Units 1-4, Tokai Unit 2 and Onagawa Units 1-3. Consequences: All ten plants survived the event, but with loss of some degree of emergency capabilities and safety margin; full recovery hampered by regional infrastructure devastation.	As above, extensive evaluations. Further opportunities to use lessons learned. All Japanese plants currently are in cold shutdown pending government decision on restart.

2.5 Acting on Nuclear Lessons Learned

Individually and collectively, the events summarized in the foregoing table contributed to better understanding of nuclear technology and safety, prompting improvements in plant and equipment design, the human-machine interface, operator training, practices and procedures, regulation, and emergency management. Underlying the learning that has occurred in all of these areas has been a steady improvement in the intangibles that influence safety within the broad umbrella of "safety culture" and ever-higher standards for safety-related aspects of plant performance.

Notably, influential learning experiences are not limited to nuclear accidents. For example, several events classified as "near misses" are included in the table above and one—the September 11, 2001 Terrorist Attacks—did not involve nuclear plants at all. Nevertheless, such events raised regulator and owner awareness of potential nuclear plant vulnerabilities to unusual or extreme events with a potential for broad area impact, and resulted in additional plant safety features.

While the nuclear safety lessons learned from operating experience cover all aspects of nuclear plant design and operation, several areas are particularly important, and are summarized here.

Reactivity Control. Power reactors harness large quantities of energy in relatively small volumes and have the potential for rapid increases in core power level. Therefore, they require highly reliable, fast-response instrumentation and controls. The SL-1 accident and the Salem near-miss were both reactivity control events, and the prompt criticality at Chernobyl was an extreme example of the consequences of inadequate reactivity control.

Reactor Core Cooling. Operating reactor cores generate a great deal of heat, which decreases but does not fully stop upon shutdown. Residual heat removal and/or emergency core cooling systems (ECCS) provide the cooling needed after reactor shutdown to prevent core damage. The TMI-2 accident and the Fukushima Dai-ichi accident both involved core-cooling-system failures, albeit from completely different causes. Maintenance of this most fundamental safety function under a broad range of initiating events, both internal and external, is of special importance in preventing future reactor accidents.

Reactor Containment. Although safety features such as ECCS are intended to prevent or arrest release of radioactivity, it is possible for accidents to progress to the point that a robust containment structure is needed to protect the public and environs. The TMI-2 containment structure effectively prevented off-site consequences, despite extensive core melting. There was no such containment in place at Chernobyl—resulting in a large release of radioactivity to the environment. The failure of containment structures at Fukushima Dai-ichi allowed a large radiological release to the environment that caused significant contamination.

Accident Management. Timely, proper, operator actions are pivotal to success in interdicting core damage during those rare circumstances that depart from normal operations. Development of sound and effective accident management processes, training, and procedures has been a key learning experience from essentially all significant nuclear-safety events. Accident management has been a factor in all major reactor accidents.

Human Performance. The human element is always a factor in reactor safety. To some extent, each of the tabulated events was influenced—positively or negatively—by the qualifications, training, practices, procedures, decisions, and overall safety consciousness of plant operators and supporting personnel.

Emergency Preparedness. In a serious accident, well-planned and practiced actions by plant personnel and effective engagement with civil authorities are key to effective public protection. The TMI-2 accident pointed out flaws in this area. Post-earthquake and post-tsunami conditions at

Fukushima Dai-ichi were unprecedented and created huge emergency management obstacles that severely challenged Japan's hierarchical process for making management decisions. The consequences of the Fukushima accident emphasize the importance of effective emergency response, even in the face of overwhelming challenges.

2.6 Summary Comments

As with the large-scale deployment of most new technologies, nuclear technology has grown safer and more reliable over the years. In all cases, experience gained from actual operations—including accidents and near-misses—has been a major contributor to this managed maturation process.

The extensive release of radioactivity to the environment with its consequential land contamination, and the ongoing challenge of cleanup and re-habitation from the events at Fukushima provide a sharp new focus on aspects of nuclear safety not fully recognized before—the large socio-political and economic impact.

Consistent with maturation of other major technologies, the learning experience from operating plants has profound effects on the design of future nuclear plants. For decades, nuclear industry organizations have been collaborating in design requirements and design concepts for the next generation of nuclear power plants. The NRC has already certified next generation plant designs and is considering others that are founded on lessons learned in the half-century of nuclear power operations in the U.S. and around the world.

For both existing and future plants, there must be a continued strengthening of nuclear plant design, regulation, and operation, with safety features that take into account both normal and accident conditions. Safety improvements must target further reductions in the likelihood of severe events and improvements in the capability to mitigate their consequences.

3 GOING BEYOND THE DESIGN BASIS

3.1 Introduction

One of the most significant components of the technical and regulatory discussions surrounding the Fukushima Dai-ichi accident is the design basis for nuclear power plants and its role in ensuring adequate protection under the law. The focus of the discussions is on what should be done to address events that create conditions beyond the design basis, such as those created by the Great East Japan Earthquake and Tsunami, where the design basis for multiple nuclear units was inadequate to accommodate the tsunami. [16][17][18] This chapter reviews the principal features in the current approach to assuring safe designs of nuclear power plants, including historically-significant transformations in design requirements, many of which resulted from the lessons learned from reactor accidents or other significant events. It also lists some of the changes proposed by others to extend the design basis as a result of the accident at Fukushima Dai-ichi. Although focused on the U.S. approach to design requirements and the importance of the design basis, this chapter is applicable to most of the global nuclear fleet, because the majority of the light water reactors around the world used the so-called "country of origin" licensing requirements, which were based on or similar to those of the U.S. NRC framework. There is an important difference between the U.S. approach to safety assurance and that followed in Europe, where nuclear power plants are reviewed every 10 years to ensure continued adequacy of their safety provisions. The NRC framework is based on the premise that plants continuously meet safety requirements as a condition for operation.

The nuclear power industry began in 1954 when the U.S. Congress passed amendments to the Atomic Energy Act (the Act), "to encourage widespread participation in the development and utilization of atomic energy for peaceful purposes to the maximum extent consistent with the common defense and security and with the health and safety of the public." [19] Later, another change in the Act charged the Atomic Energy Commission (AEC), now the Nuclear Regulatory Commission (NRC or Commission), with responsibility for ensuring protection of public health, safety and the environment from hazards associated with industrial applications of atomic energy. A guiding statute for the Commission's responsibilities in the area of health and safety of the public is Section 161b of the Atomic Energy Act; i.e., the Commission is to "establish by rule, regulation, or order, such standards and instructions to govern the possession and use of special nuclear material, source material, and byproduct material as the Commission may deem necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property." Current regulations and actions of the Commission are usually within the framework of "adequate protection," i.e., the Commission's actions "will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public." Rules and regulations from the NRC are commonly enacted under the terms "reasonable assurance of adequate protection," or under "no undue risk to the public health and safety," or "no imminent risk to public health and safety." [20]

The subject of adequate protection has been reviewed by Congress, the Commission, multiple organizations and stakeholders, and the courts. A landmark ruling was issued in 1987 by the Court of Appeals for the District of Columbia Circuit. The Court ruled, "The level of adequate protection need not, and almost certainly will not, be the level of zero risk." In the ruling, the Court of Appeals acknowledged that use of the word "adequate" by the Act implied some degree of discretion on the part of the NRC. The Court also stated that "adequate protection" did not mean nuclear energy had to be entirely risk-free; rather, a certain amount of risk was acceptable. The ruling is, of course, compatible with the reality that there is no such thing as zero risk, and for all technologies, including nuclear, a certain level of risk is acceptable to society. [1]

It can be stated that nuclear power has become a strictly-regulated energy production method, in many ways significantly surpassing the regulatory structures for other technologies, with a clear delineation of authority and responsibility between government and industry. That is, government entities (Congress and NRC) set the expectations for nuclear safety in the commercial sector while industry participants (plant suppliers, owners, and operators) are primarily responsible for achieving safety in nuclear power plant design and operations.

Based on this framework, government and industry have managed radiological risk by providing features in the design of nuclear power plants to both prevent and mitigate accidental release of radioactive material to the environment. Over the years, such design features have been improved to reflect advancements in technology, changes in socio-political factors, expectation of higher levels of safety for newer plants, and accumulation of operating experience. Improvements continue to be made in consonance with advancements in science, operating experience, and public policy.

3.2 How Designs Have Been Established Up to Now

3.2.1 The Design Basis

The primary safety requirements that govern the design of nuclear power plants in the U.S. are contained in Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants." The safety principles embodied in these general design criteria are widely emulated in nuclear power plants around the world. Other sections of 10 CFR 50 provide more specific criteria to implement the general design criteria.

The design basis for each nuclear power plant is described in a Safety Analysis Report (SAR) that supports the operating license for that plant. The technical information contained in such reports includes the seismic, meteorological, hydrologic, and geologic characteristics of the plant site, "with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated." The SAR also contains safety assessments of the performance of the plant under a specified set of conditions that define a design envelope. All of those safety assessments are required to show that the reactor core is maintained in a coolable geometry and that there would be limited releases of fission products to the reactor coolant system or the reactor containment. In addition, the combination of site characteristics, demography, and plant design must be shown to meet certain radiation exposure criteria for offsite individuals in the event of an arbitrarily-large release of radioactive material from the reactor coolant system into the reactor containment.

The design envelope described in the SAR includes conditions (structural loads, temperatures, pressures, life expectancy, radiation levels, etc.) representative of

- Normal operations (e.g., power production, shutdown, refueling),
- Anticipated operational occurrences (e.g., reactor trip, water hammer),
- Infrequent events or transients (e.g., loss of offsite power, loss of feedwater),
- Design basis events (e.g., floods, earthquakes, external man-made hazards), and
- Design basis accidents (e.g., loss of coolant, large reactivity changes).

The design basis includes accident prevention and accident mitigation requirements for each nuclear power plant, for a defined scope of events. The requirements for all events within the design basis are that the fission process can be shut down, the reactor remains subcritical, the core can be cooled and its geometry maintained, coolant makeup can be supplied as needed, and, in the event of an accident involving core damage, any fission products released from the core can be controlled within a containment structure that assures limited radioactivity leakage to the environment. Information contained within the design basis for a particular piece of equipment or building includes, for example, the specific functions to be performed and the specific values or ranges of values for conditions that serve as the bounds for its design. These conditions are derived from engineering analysis or experiments to predict the consequences of events postulated for design, including those in the preceding list. An important lesson learned from decades of operating nuclear power plants is that rigorous configuration control must be applied, throughout a plant's operating life, to assure that the design basis is maintained.

The transformative event associated with the accident at TMI-2 was the realization there was another segment of the design basis that had not been fully explored, i.e., accident interdiction. This term, interdiction, means interruption of an accident scenario as it progresses from an upset condition within the design basis to more serious conditions outside the design basis, which involve fuel heatup, cladding swelling and rupture, and autocatalytic cladding oxidation leading to release of fission products from the fuel and core melting. At TMI-2 (and Fukushima Dai-ichi) there was a relatively long period of time (hours to a few days) between failure of safety equipment to prevent onset of core damage and the need to mitigate radiological consequences of core damage. This period allows for use of other design features to interdict an accident, short of a complete core meltdown and breach of the primary coolant system if operators are trained to use these design features and the features remain available during the accident progression. The possibility of interdicting an accident in progress was an important lesson learned from the TMI-2 accident and came to be known as "accident management," as discussed in Chapter 6.

Until the events unfolded at Fukushima Dai-ichi, a common assumption was that accident management involved making the best use of available equipment—whatever its status—to arrest core damage, before it led to large offsite releases of radioactivity. Unfortunately, because of the extended loss of all electrical power, there was little stationary equipment available at Fukushima Dai-ichi. However, those events highlighted the potential efficacy of safety improvements made at U.S. nuclear power plants after the September 11, 2001 terrorist attacks, which were largely absent at Fukushima. The improvements were addition of design features—or supplemental equipment—specifically tailored to interdict the progression of an accident, even under extreme conditions, including loss of electrical power. Due to the complicating factors of hydrogen generation and high radiation levels in the plant—as exhibited at Fukushima Dai-ichi—interdiction to prevent extensive core damage is of the highest priority, with mitigation of core damage consequences providing defense-in-depth. Seen in this light, interdiction of core damage occurs in a transition regime between prevention of core damage and mitigation of its consequences.

The U.S. NRC is empowered to implement regulations and promote safety programs to ensure that nuclear power plants are operated with reasonable assurance of adequate protection. Practical use of the NRC's adequate-protection standard has evolved over the years, primarily based on operating experience. The engineering basis of adequate protection is stated in the introduction of Appendix A to 10 CFR 50 as, "the facility can be operated without undue risk to the health and safety of the public." The Introduction also states that the set of General Design Criteria contained in Appendix A "is not yet complete." Because some of the incomplete parts of Appendix A were related to the accidents at TMI-2 and Fukushima Dai-ichi (e.g., common-cause failures and combinations of events) this situation should be corrected. Reforms aimed at some of these factors are under consideration by the NRC and the nuclear industry, as described below. Others deemed necessary to complete Appendix A should also be undertaken, including risk-informing pertinent criteria.

Through the years, the NRC has expanded the design basis for safety reasons and to correct deficiencies. Also, science and technology have provided many opportunities to reduce unnecessary conservatisms. New knowledge and experience combine to provide the rationale for change. Moreover, improvements in areas outside regulatory compliance are often covered by voluntary industry programs that are accepted by the NRC for implementation at licensed facilities. Examples of events that first were judged to be outside the design basis and then were later regulated by the NRC include anticipated transients without scram, loss of all AC power (station blackout), design features for hydrogen mitigation, and coping capability for large external fires or explosions. The events that unfolded at Fukushima Dai-ichi have led others to suggest a number of additional design measures as candidates for extending the design basis, as discussed below. Some of these design measures have already been incorporated into the new Generation III+ designs for light water reactor plants, one of which has now completed regulatory review for design certification; i.e., licenses have been granted and construction has started on four sites referencing the Westinghouse **AP1000**®[2] PWR Design Certification, including two sites in the U.S. [21]

In the past, events that were judged to be beyond the design basis were given various names. Before the accident at TMI-2, they were called class 9 accidents and were judged to be out of bounds for environmental impact statements (EISs) associated with licensing new plants because their probability was considered to be too small. After the partial core meltdown that occurred at TMI-2, the NRC abandoned the class 9 terminology and began to consider the consequences of severe accidents in EISs. In that same time frame, the NRC also began to require that severe accidents be considered in design of plants still on the drawing board, including the Generation III and III+ plants. Changes in the scope of the design basis continued when the NRC invoked risk insights elsewhere in the regulatory process, e.g., the requirement to "risk inform" maintenance programs for operating plants. Accidents beyond the design basis now have become the subject of extensive probabilistic risk assessments (PRAs), as discussed more fully in Section 3.2.4.

In the 1980s, while the NRC was expanding its consideration of accidents beyond the design basis, it also undertook efforts to define acceptable risk, i.e., to define how safe is safe enough for nuclear power. That was a difficult undertaking, because societies do not have a universal measure of what constitutes acceptable risk. Instead, risk acceptance varies with the perceived importance of the activity, whether the risk is undertaken voluntarily or not, and its history, the trust in the technology and its purveyors, and the regulatory or legal framework. This variability of risk acceptance is well known and is unlikely to change. [22] Thus, people accept some everyday risks (having a car accident), ignore others (residing in areas susceptible to large earthquakes or floods), take some voluntarily (smoking), and reject others (jumping off a cliff). Public reactions to the events at Fukushima Dai-ichi and to other major disasters indicate that the involuntary risks of living in modern society are tolerated until they are no longer perceived to be safe or environmentally acceptable, in some cases somewhat irrespective of their actual potential for harm.

For nuclear power, previous attempts to define acceptable levels of risk have been based on the potential health consequences, i.e., prompt fatality risk or latent cancer fatality risk. Relative risk also has been suggested as a measure of acceptability, e.g., comparing deaths from lightning strikes with death from radiation-induced cancer, or multiplying probability times consequences as an indicator of risk. However, because of the inevitability of human errors and the inherent unpredictability of nature, rare events occur that exceed expectations. It is a daunting task to define acceptable risk in a manner that accommodates uncertainties in the small but finite probability of high-consequence rare events.

Despite such difficulties, in 1988, following the accidents at TMI-2 and Chernobyl, the NRC published a policy statement on safety goals to aid its judgments about what constitutes acceptable

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risk for nuclear power plants. [23] The two goals contained in the statement address public health risk for individuals and society based on comparisons with risks from other causes. That is, NRC stated that the public health consequences of nuclear power plant operation should be one thousand times less likely than similar consequences arising from other causes, and the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation. The Commission also noted, "Apart from their health and safety consequences, severe core damage accidents can erode public confidence in the safety of nuclear power and can lead to further instability and unpredictability for the nuclear industry."

A decade later, the IAEA advanced the establishment of safety goals for nuclear power, including a target frequency of occurrence of severe core damage that is below about 10^{-4} per year for existing plants. [24] The IAEA said it expected that accident management and mitigation measures could reduce the probability of large off-site releases by a factor of at least ten. The IAEA also expected future plants to achieve an improved goal of not more than 10^{-5} severe core damage events per year, and thus practically eliminate "accident sequences that could lead to large early radioactive releases."

The EPRI worked with representatives of a dozen countries to develop a Utility Requirements Document for advanced light water reactors, which led to new and safer reactor designs now being deployed. [25] The EPRI program uses a concept it calls investment protection, to set the core damage requirement of less than 10^{-5} core damage events per year and a large release probability of less than 10^{-6} events per year. The program also defined the capabilities required of Generation III+ plants with passive safety features, e.g., assure adequate core cooling without AC power or operator action, for at least 72 hours.

Nuclear operations worldwide (including the events at Fukushima Dai-ichi) have demonstrated the industry's ability to meet or exceed such safety goals from the perspective of public health and safety. However, there is work to be done from the perspective of preventing severe accidents with large releases of radioactivity to the environment and the high socio-political and economic costs they entail. It has been suggested that avoidance of these consequences is critical to acceptability of nuclear power. For example, within weeks after the events at Fukushima Dai-ichi, an ad hoc international group of senior nuclear safety specialists from a dozen countries advised the Director General of IAEA, "only nuclear power that avoids being a threat to the health and safety of the population and to the environment is acceptable to society." Achievement of that goal requires that the nuclear power industry redouble its efforts to ensure there are no more accidents with such large offsite releases of radioactivity, like Chernobyl and Fukushima. [26]

Because there is broad-based sentiment for further improving capability of nuclear power plants to address severe external events, an important question arises as to what the basis should be for changes to design and accident management. In 1988, to establish limitations to issuance of new regulations, the NRC issued a backfit rule (10 CFR 50.109) to describe the circumstances under which it would, among other things, require "modification of or addition to systems, structures, components, or design of a facility [i.e., require a modification of the design basis]." [27] The backfit rule addresses how the Commission makes its determinations of what is required for adequate protection and its determinations of what constitutes mere "safety enhancements." The costs to be considered in a backfit analysis are described in NRC and industry guidance documents, which discuss the need to address not only health effects but also offsite land contamination and adverse effects on the "efficient functioning of the economy." [28]

The new safety construct recommended herein rises above the immediate debate of what is required for adequate protection of public health and safety under NRC's enabling legislation or whether changes under consideration are justifiable under NRC's backfit rule. Instead, it is envisioned that the new safety construct must provide an expected outcome with a clear purpose, for the totality of safety measures embraced by the nuclear industry to ensure its present and future acceptability to society in general. The intent is to provide a framework for an all-risk approach in a transparent manner where the goal is clear, the social risks are well defined, and the overall, integrated effect of protective safety measures is well understood. The costs of developing and implementing such a framework are considered by the ASME Task Force to be small in two respects: (1) in relation to the unacceptable socio-political and economic costs of events that lead to widespread radioactive contamination of the environment; and (2) in relation to the potential for lost economic benefits if existing plant operation and growth of the industry are impacted by lack of social acceptance and public trust.

3.2.2 Defense-In-Depth

As described in Section 3.2.1, the design basis requirements for nuclear power plants are mostly embodied in 10 CFR 50, Appendix A, which has become the basis for nuclear safety around the world. The quality requirements for the structures, systems, and components of Appendix A are in 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Besides the design basis, there is another fundamental safety principle embedded in Appendices A and B, namely, defense-in-depth.

Defense-in-depth has been implemented in the design basis of nuclear power plants since the earliest demonstration reactors in the 1960s, but there is no regulatory definition of defense-in-depth. Rather, it "is considered to be a concept, an approach, a principle or a philosophy, as opposed to being a regulatory requirement per se." [29] Simply stated, the principle of defense-in-depth requires four levels of protection of public health and safety to be present, as follows:

1. Nuclear power plants are designed to not fail for the range of normal and abnormal conditions they might reasonably experience during their operating life, including extreme natural phenomena and man-made hazards;
2. Redundant and diverse support systems are provided to detect initiating events or incipient failures and shut the reactor down before fuel damage can occur following events within the design basis;
3. Redundant and diverse emergency systems, including emergency core cooling and containment systems, are provided to limit the release of radioactivity to the environment and thereby mitigate accident consequences if the first two levels of defense fail; and
4. Reactors are sited remotely and emergency plans are prepared and practiced in advance with local and regional authorities to limit the public consequences of a catastrophic accident involving a large release of radioactivity.

Another way to look at defense-in-depth is that it is like a four-legged stool, where all four legs listed above are required to be in place whenever a nuclear power plant is operated. Whenever a leg is out of service, a plant is required to shut down until all four are in place. The NRC has acknowledged that defense-in-depth includes measures beyond the design basis, as follows: [30]

> "The concept of 'defense-in-depth' is a centerpiece of [the nuclear industry and regulatory authorities'] approach to ensuring public health and safety, and it goes beyond pieces of equipment. It calls for, among other things, high quality design, fabrication, construction, inspection, and testing; plus multiple barriers to fission product release; plus redundancy and diversity in safety equipment; plus procedures and strategies; and lastly, emergency preparedness, which includes coordination with local authorities, sheltering, evacuation, and/or administration of prophylactics (for example, potassium iodide tablets). This approach addresses the expected as well as the unexpected; it actually accommodates the possibility of failures. The NRC's defense-in-

> depth has recently been strengthened by incorporating the dynamics of risk-informed and performance-based decision making."

The recent addition of numerical risk insights to the principle of defense-in-depth complements the deterministic approach to the design basis. The risk insights are gained through PRAs. A basic conclusion developed by the nuclear industry over the past three decades is that defense-in-depth is necessary—but not sufficient—to ensure adequate protection, i.e., unless you inform a design by PRA, you cannot conclude it is adequate. To carry that conclusion a step further, in light of the events at Fukushima Dai-ichi, the ASME Task Force considers that it is essential to implement an all-risk approach that would support avoidance of the socio-political and economic cost of severe accidents, including use of full-scope PRAs, as described in Section 3.2.4.

3.2.3 Deterministic Approach to Achieve Defense-In-Depth

In the deterministic approach, the design basis for a feature (structure, system, or component) in a nuclear power plant is defined by an analysis of its effectiveness for the conditions it is intended to control or mitigate. The conditions to be addressed in the design, the methods of analysis, and the acceptance criteria are specified in advance by regulatory authorities for safety features (e.g., emergency shutdown and cooling of the reactor) and by the owner of the plant for non-safety-related equipment. In the U.S., this historical distinction between the regulator and the owner's scopes of interest has blurred in the review of Generation III and III+ designs, which are done under the requirements of 10 CFR 52.

For plants now in operation, the design basis accidents and events specified in the deterministic approach used in their design generally involved

- Single initiating events,
- Conservative assumptions and models,
- Aggravating single failures and loss of offsite power, and
- No expectation of severe core damage.

Exceptions to this general approach involve combining some accidents with severe natural phenomena (e.g., the design basis loss-of-coolant accident (LOCA) is combined with the design basis earthquake) and situations where realism is included in the analysis to aid understanding of the phenomena involved. Because the acceptance criteria for design basis accidents allow only limited or no fuel damage, the containment designs for plants now in operation were set deterministically, and to some degree independently of the outcome of design basis accidents. Thus, the design for containments like the ones at Fukushima Dai-ichi included

- Design pressure associated with successful mitigation of the design basis LOCA,
- Leakage requirements based on release of a substantial fraction of the fission products in the core to the containment,
- Venting capability associated with removal of decay heat and other heat sources;

There were no design provisions for a molten core that penetrated the reactor vessel.

New designs licensed under 10 CFR 52 require provisions to address severe core melting and potential penetration of the molten core materials (corium) through the reactor vessel. Typically, there are four provisions:

1. Cool the corium by spreading and flooding (or retain the molten core in the reactor pressure vessel by an in-vessel retention system which cools the outside of the vessel).

2. Burn hydrogen as it is generated.
3. Minimize the core-concrete interaction that could generate gases that would contribute to pressurization of the containment.
4. Prevent the molten corium from spreading to the upper containment where it could produce local hot spots and cause localized over stressing.

Engineers provide design margins in the deterministic approach to nuclear power plant design, much like engineers provide margins in other designs, such as bridges and airplanes. A design margin is the distance between the bounding prediction of a load or other condition and the point at which the potential for failure due to that condition becomes non-negligible. Design margins, usually called safety margins when discussing specific nuclear safety-related issues, help account for uncertainties and unknowns, as well as wear and tear, e.g., corrosion or cyclic fatigue of a pipe.

In the deterministic approach to design of nuclear power plants, safety margins are included in selection of design methods, design criteria, codes and standards, and operating limits. Equally important, operating conditions are kept within limits by an amount commensurate with the uncertainties involved in setting the limit and measuring the operating conditions.

Operating experience has revealed some limitations in the deterministic approach to the design basis, particularly in the areas of common-cause failures and human error. One example was the failure of the deterministic approach in the design of TMI-2 to sufficiently account for operator error in interpreting readings from the level instrumentation for a leak high in the pressurizer of the reactor, leading the operators to prematurely terminate emergency core cooling. Another example was the inability of the single failure criterion, widely used in the deterministic design approach, to anticipate the risk associated with maintenance errors that caused common-cause failures in the reactor scram system at the Salem Nuclear Power Plant.

Another limitation of the deterministic approach is its high reliance on conservative analysis of the course and consequences of design basis accidents. Overly conservative analyses can simplify the prediction of accident progressions to the point that they hide phenomena that occur along the way. For example, post-accident review by the NRC and the nuclear industry showed that more realistic analyses of LOCAs and high-pressure core-melt sequences, and training on the interpretation of the results of such analyses, would have aided the TMI-2 operators in diagnosing the accident underway in their plant and would have improved their chances of interdicting that accident short of core melt or their chances of further limiting the extent of core melting. [31]

3.2.4 Probabilistic Approach to Achieve Defense-In-Depth

In the 1970s, as the nuclear industry was gaining operating experience, there was a need to characterize the risks associated with the growing number of power plants. The AEC performed a probabilistic study of the safety of two typical plants then operating. The idea was to compare the risks from those typical plants to other risks posed to society by technology and natural phenomena. The study was known as the Reactor Safety Study. [32] It contained the first PRAs of nuclear power plants and identified several strengths and weaknesses of the deterministic approach to design.

- The most important contributors to nuclear power risk are not the large LOCAs that underlie most safety system design requirements in the currently operating plants. Rather, the dominant contributions to risk come from combinations of more-likely events, e.g., small LOCAs and loss of offsite power, along with accidents involving coincident or cascading failures of multiple components or systems. On one hand, this result was a credit to the deterministic design approach. Conversely, this result indicated that focusing solely on limiting events might not ensure adequate safety.

- The important role of operator actions was also brought to light by the Reactor Safety Study, especially time-critical operator actions, a fact that was underscored by the accident at TMI-2 four years later.
- There is a substantial contribution to risk from multiple failures due to a single cause, i.e., common-cause failures, which were not considered in the design of existing plants and were only addressed in the deterministic framework through programmatic controls.

Although PRA was not used in the design of the currently operating U.S. nuclear power plants, it has been used for license amendments and to improve decision making for those plants, including design and operational improvements to prevent, interdict, and mitigate accidents beyond the design basis. As discussed in Section 3.4, PRA has been an integral and required part of the design process for the Generation III and III+ plants, searching for vulnerabilities and correcting them through design changes, assessing response to accidents beyond the design basis, and defining the level of safety achieved.

The PRAs in the Reactor Safety Study and those that followed take a fundamentally different approach to the assessment of safety than the deterministic approach. At its core, PRA attempts to answer three fundamental questions through development of an integrated model of the as-built, as-operated plant (as well as new plant designs), namely, what can go wrong, how likely is it to occur, and what are the consequences?

The PRA approach is more comprehensive than traditional deterministic approaches in addressing accident scenarios, causes of system and human failures, and treatment of uncertainties, as outlined in Table 2 [33].

Table 2 – Comparison of Deterministic and PRA Approaches to Safety Assessment

Consideration	Deterministic Approach	PRA Approach
Scope of Events Analyzed	• Pre-defined set of events • Assumes design basis events are bounding	• Not constrained by pre-defined rules
Failure Scenarios Included	• Worst single active failure assumed to occur	• Unlimited number of failures considered probabilistically
Common-Cause Failures	• Assumed to be precluded by special treatment requirements	• Probabilistically considered for all equipment based on experience
Human Actions	• Assumed effective when proceduralized	• Human actions, both positive and negative, are considered probabilistically
Approach to Uncertainties	• Dependent upon bounding assumptions	• Focus on mean (realistic) estimates and quantitatively assess uncertainties

Although the fortunes of PRA have waxed and waned (e.g., Congress forbade its application in licensing actions by the NRC until 1980 when the TMI-2 accident sequence was shown to be one of the dominant risk contributors in the Reactor Safety Study), PRA now has become a valuable adjunct to the deterministic approach. Through the 1980s, the NRC used PRA to investigate various issues coming from the TMI-2 accident, and in 1988 the NRC requested that all plants perform a plant-specific PRA under the Individual Plant Examination (IPE) program. [34]

The industry and NRC gained significant safety insights from the IPEs, and by the mid-1990s use of PRA had become quite common in support of operational and licensing decisions. In 1995, the NRC issued a policy statement that combined principles of defense-in-depth, deterministic approach to the design basis, and use of PRA, going so far as to say, [35]

> "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."

The so-called "risk-informed" approach to regulatory decision making is embodied in NRC's Regulatory Guide 1.174 and now permeates many aspects of regulatory activities, including [36]

- Operational decisions related to equipment removal from service for maintenance,
- Optimization of inservice inspection and testing programs,
- Prioritization of NRC and licensee activities,
- Development and approval of license amendments,
- Refinements to technical specifications governing plant operations, and
- Assessment of the significance of performance deficiencies identified by NRC.

Risk assessment is now recognized as the best available method for identifying and addressing uncertainties in safety decisions based on predictions of plant performance under abnormal and extreme conditions. Moreover, integration of PRA results with deterministic defense-in-depth considerations, as described in NRC's Regulatory Guide 1.174 and IAEA's Safety Guide INSAG-25, [37] will yield even more robust safety decisions. Based on review of past accidents and seeing the utility of PRAs in forecasting the outcome of other rare events, the ASME Task Force considers that all-modes, all-risk, full-scope risk assessments, including level 3 (consequence) analysis, should be combined with deterministic approaches, to achieve greater defense-in-depth for all nuclear power plants. A similar recommendation was recently made by an NRC Risk Management Task Force, led by Commissioner George Apostolakis, which said that full scope PRAs are the preferred tool for measuring safety and they should be applied wherever practical. [38] Furthermore, the ASME Task Force considers that there should be a continuing international effort to improve ways to integrate risk assessments with the deterministic, defense-in-depth approaches to design of nuclear power plants. The recent joint effort by the ASME and the ANS is a substantial step in that direction. [39]

In addition, to take maximum advantage of such improvements in state-of-the-art of risk assessment, the ASME Task Force considers that generic, high-level safety goals for new plants should be agreed internationally, with the aim of reducing the probabilities of core damage accidents and limiting radioactive releases to the environment. However, even PRA is not all knowing. Therefore, an all-risk approach is needed to turn the question around so that engineers provide systems and actions to ensure core cooling and prevent large releases of radioactivity for any rare yet credible event. In the words of Prime Minister Yoshihiko Noda on the first anniversary of the Great East Japan Earthquake and Tsunami, "Crisis management requires us to imagine what may be outside our imagination." [40] Such a consideration is important for assembling the new safety construct.

There are two important insights related to the probabilistic aspects of defense-in-depth from the events at Fukushima Dai-ichi that relate to the new safety construct. The first is that beyond-design-basis events can occur. It is not sufficient to assume that designing and operating a plant to its design basis will ensure avoidance of an accident. If the uncertainties are large, additional interdiction and mitigation capability must be provided. This is particularly true of events with the potential for so-called cliff-edge effects.

Second, PRA is an effective tool in identifying and developing coping measures for plant-specific conditions that exceed the design basis. For example, PRAs for a particular site can evaluate the risks from rare yet credible natural events, such as large earthquakes and tsunamis that could occur, either individually or in combination. Such assessments, when performed in conjunction with deterministic evaluations of the adequacy of protective measures for rare yet credible events, lead to informed determinations of whether these potential events would be dominant risk contributors at those sites. Moreover, such assessments lead to identification of potential core damage scenarios and accident management and interdiction measures that can be taken or planned in advance. Such proactive steps can substantially alter the course of low-probability, high-consequence events, thereby preventing damage to the reactor fuel and the subsequent release of radioactivity to the environment. This approach already has been taken in many nations and should be taken by all, to identify gaps in defense-in-depth caused by incomplete assessment of plant risks provided by the traditional deterministic approach.

3.3 How Designs Might Change to Reflect Lessons Learned from Fukushima Dai-ichi – The Emergent Safety Construct

The degradation of safety performance that occurred in Units 1- 4 of the Fukushima Dai-ichi plant in the wake of the Great East Japan Earthquake and Tsunami indicates that pertinent complements to the design, which could depend on fixed and mobile equipment, would be beneficial in currently operating nuclear power plants. The overall purpose of such design changes is to reduce the risk associated with extreme external events and to improve the capability to cope with accidents beyond the design basis. That is, the prior safety construct for nuclear power plants should be improved given our present understanding and ability to analyze the potentially large consequences of severe external events.

One of the important lessons learned from the events at Fukushima is that the likelihood and severity of rare natural phenomena are hard to predict. Consequently, it cannot be sufficient to simply revisit the traditional design basis as a means to protect nuclear power plants from fuel damage and prevent large offsite releases of radioactivity. Although the ability to predict the magnitude and frequency of natural phenomena such as earthquakes and floods might be improving, significant levels of uncertainty will always remain. Additional steps to address the effects of the beyond-design-basis events from severe natural phenomena would enhance safety at each site.

In Europe and Japan, design changes for reduction of risk due to severe external events are being evaluated through plant-specific reassessments of the events, features, and processes that have been taken into account in the design basis for both new and existing plants, including facilities for storing used fuel. These reassessments are called “stress tests,” and are now completed in Europe and still underway in Japan. In the U.S., the NRC and industry are following a similar course based on recommendations of the NRC’s Near Term Task Force on the Fukushima Dai-ichi Accident and efforts by the nuclear industry to make improvements addressing Fukushima events. [5] The NRC Task Force found, among other things,

> “…that the Commission’s longstanding defense-in-depth philosophy, supported and modified as necessary by state-of-the-art probabilistic risk assessment techniques, should continue to serve as the primary organizing principle of its regulatory framework. The [NRC] Task Force concludes that the application of the defense-in-depth philosophy can be strengthened by including explicit requirements for beyond-design-basis events…. The [NRC] Task Force has

> concluded that a collection of…'extended design-basis' requirements, with an appropriate set of quality or special treatment standards, should be established."

The NRC's specific, design-related actions in response to the events at Fukushima Dai-ichi involve some short term changes to be issued by orders without the formality of rulemaking and some longer term issues to be accomplished through rulemaking. These changes include consideration of design-related improvements for operating and new plants like those listed in Table 3. [41]

Table 3 – Improvements Suggested by Events at Fukushima – The Emergent Safety Construct

Prevention	Interdiction	Mitigation
Design for Combined Earthquake & Tsunami	Better Coping Capability for Loss of All AC Power	Hardened Vents for Various Containment Designs
Design for Larger Tsunamis and Other Extreme External Flooding	Account for Possible Loss of Infrastructure (transportation, communications, etc.)	Hydrogen Burners in and out of Containment
Improved Spent Fuel Cooling	Better Coping Equipment for Large Fires and Explosions	Contaminated Water Storage
Improve Designs for Internal Flooding	Alternative Core Cooling Paths	Improve Command and Control Centers
Design for Earthquake + Fire & Earthquake + Flooding	Alternative Power Connections	Improve Emergency Data Acquisition and Transmission
Better Ultimate Heat Sinks	Alternative Pumping Capability & Water Supply	Filtration of Containment Vents or Comparable Measures

On December 15, 2011, the NRC Commissioners issued directions to their staff on "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned." [42] The Commission chose to consider some of the new requirements under its backfit rule rather than to adjust its prior interpretations of the adequate protection standard. Presumably, such use of the backfit rule will include consideration of the full cost of offsite releases of radioactivity in deciding whether the costs and benefits of accident interdiction and mitigation measures are justified. It is important to realize that if systematic consideration is to be given to preventing significant societal consequences, primarily due to offsite radiological contamination, as proposed herein for the new safety construct, a complete valuation of additional design features would eventually require the performance of full scope PRAs for each nuclear power plant, i.e., all-modes, all-risks, full-scope risk assessments, including level 3 (consequence) analysis.

On February 17, 2012, the NRC Staff provided for Commission consideration proposed orders encompassing needed short-term enhancements of safety indicated by the events at Fukushima Dai-ichi. [43] The majority of the Commissioners approved these orders in late February 2012.

Concurrently, the U.S. industry has proposed to implement an added accident management capability at all operating plants, which has been named "FLEX" and is described in Section 6.6.

These efforts by the NRC and the industry to make changes related to the events at Fukushima Dai-ichi will be important pillars of the New Nuclear Safety Construct described in Chapter 1. However, the Commission has not dealt with the NRC's Near Term Task Force recommendation for "establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations." The NRC's Executive Director for Operations has said the NRC staff will provide the Commission with options and recommendations by March 2013 on how to proceed with that recommendation. [44] Thus, at the time of this writing, no complete safety framework has emerged from the NRC as a result of the events at Fukushima Dai-ichi. However, both the NRC and U.S. industry efforts are sound complements to the existing safety framework and provide a better approach to addressing a number of issues raised by the accident. Cohesiveness, coordination, and completeness should be added in the form of a definitive overarching purpose for such improvements, as would be provided by the New Nuclear Safety Construct recommended herein.

3.4 Designing New Nuclear Power Plants

The designers of new nuclear power plants have already incorporated safety improvements that accommodate some of the lessons learned from the events at Fukushima Dai-ichi, such as improved reliability of core cooling in the absence of AC power. They also have given increased consideration to multiple failures and to accidents that are beyond the design bases for operating plants. In addition, the designs of all new plants are required to include a PRA, performed as an integral part of the design process, to inform the operational framework, plant layout, plant design, safety system capabilities, and mitigation features for severe accidents. Prior to fuel load, all new plants will be required to have a comprehensive, plant-specific PRA, addressing site-specific natural phenomena and internal hazards. New plants also feature improved design measures for accident prevention (e.g., reduced likelihood of common-mode failures, reduced complexity, increased inspectability and maintainability, more use of passive safety features, improved human-machine interface, and extended use of information technology) and design features to reduce the likelihood and consequences of off-site releases of radioactivity (e.g., combustible gas control systems, reactor cavity spreading and flooding systems, and reactor coolant system depressurization). When the new plants begin operation, there will be a need to feed back operating experience to improve the effectiveness of their innovative design features.

The NRC's regulatory requirements were modified after the accident at TMI-2, to require more thorough treatment of severe accidents in the design of future plants. Today, the NRC requires an application for design certification to include a Design Certification Environmental Report (DCER). [45] The NRC also specifies that the DCER "must address the costs and benefits of severe accident mitigation design alternatives (SAMDAs), and the bases for not incorporating severe accident mitigation design alternatives in the design to be certified." [46]

For future plants, it will be possible to do more than was done in the past to preserve and maintain the design basis across a fleet of similar plants—particularly in the expanded form of the design basis as envisaged by this paper, and already in place to a degree in Generation III+ plants. The CORDEL (Cooperation in Reactor Design Evaluation and Licensing) program of the World Nuclear Association (WNA) is working to achieve international harmonization of nuclear standards for advanced reactor designs for this and other purposes. [47] This issue could lead to international agreements on the role of the designer across international boundaries, including national regulatory authorities and other organizations, as has occurred in the aircraft industry.

3.5 Summary Comments

The design basis for operating nuclear power plants has served the nuclear industry well from a public health and safety perspective. Although the design basis for operating plants has been the foundational public health and safety strategy for the nuclear industry, the thousands of reactor-years of operating experience accumulated worldwide now provide an opportunity to improve safety beyond the design basis and, in doing so, evolve toward a new safety construct. Indeed, rare yet credible events can occur which may exceed the design basis and potentially lead to an accident with major socio-political and economic costs associated with significant radioactivity releases to the environment. No new, overarching safety construct has yet emerged for existing and future nuclear plants as a result of the Fukushima Dai-ichi experience. What appears to be needed is an objective standard for preventing, interdicting, and mitigating severe accidents, to prevent or minimize core melting and extensive offsite contamination, using an all-risk approach. Such a standard should be built upon the historically-acceptable design basis, supplemented by additional safety measures to increase the level of defense-in-depth and reduce risk.

Even though advancements have been made in safety of new nuclear power plants, research and development should continue to seek more opportunities to improve their safety. Such research might include improving fuel design; strengthening protection barriers; increasing reliability and availability of passive and active safety systems; and modeling fuel, reactor, and containment behavior in severe accident scenarios. Social sciences important to the matters discussed in Chapters 5 through 8 also deserve further research (e.g., performance of private and government organizations during emergency situations and improving human reliability), as do natural sciences (e.g., hazards associated with natural phenomena, such as floods and earthquakes.)

4 ACCIDENT PREVENTION AND CORE COOLING: THE PRINCIPAL SAFETY STRATEGY AND THE OVERRIDING SAFETY FUNCTION

4.1 Introduction

As described in Chapter 3, the safety of nuclear power plants involves defense-in-depth, so that accidents are prevented from occurring; interdicted during the progression of the accident, if possible; and mitigated, as a last resort, to minimize release of fission products. Of these three, prevention is, and has always been, the principal strategy. Thus, maintenance of core cooling is the overriding safety function.

This bears repetition. The most important safety function in a nuclear power plant is to cool the reactor core. This function persists no matter what state the nuclear reactor is in, i.e., normal operations, shut down for refueling, experiencing an upset in normal operations, recovering from an accident, or undergoing a core melt down. Whatever the condition of the reactor core, without hesitation, reactor operators must continue to cool it by whatever means possible.

This chapter reviews the core cooling measures provided for light water reactors, including historically-significant transformations in those measures, some of which resulted from lessons learned from reactor accidents or other significant events. It also lists some of the changes proposed by others to improve the capability to provide core cooling as a result of the accident at Fukushima Dai-ichi. Although the discussion that follows is focused on light-water reactors, the importance of core cooling applies to all nuclear reactor design types.

The core of a nuclear reactor is located inside a reactor pressure vessel, which is a part of the reactor coolant system (also called the primary coolant system). The core contains nuclear fuel assemblies, in which the nuclear reaction takes place and produces heat during operation. The fuel assemblies in a typical light-water reactor are comprised of fuel rods made of low-enriched uranium oxide pellets housed in sealed tubes several meters in length made of an alloy of zirconium metal. The core also contains control rods, which are used to quickly shut down the reactor when necessary or to control the amount and rate of increase or decrease of power, and various structural materials that hold the fuel and control rods and guide the flow of coolant to the fuel rods. The core also houses the detector portions of various instrumentation systems (e.g., thermocouples and neutron detectors).

In a light water reactor, splitting of uranium atoms (primarily U^{235}) forms highly radioactive fission products. Some of the fission products decay while the reactor is operating, and the thermal energy they release while decaying is removed from the core along with the heat produced by the fission process. The fission products in the reactor at the time it is shut down continue to decay and release thermal energy. The decay heat immediately following reactor shutdown is about 7% of the power level at which the reactor operated prior to shutdown. Thus, a reactor operating at 3,200 MWt will produce 224 MWt of decay heat immediately after shutdown. The decay heat decreases exponentially after shutdown, reaching about 2% of the pre-shutdown power level within the first hour after shutdown, 1% within the first day and 0.5% at 36 hours. Decay heat is still very significant, from a core-cooling perspective, for several more weeks. Failing to cool the fuel after shutdown results in fuel heatup, which could then result in fuel melting and the attendant production of large amounts of hydrogen gas, by oxidation of zirconium fuel cladding in contact with steam.

4.2 Growing Recognition of Core Cooling as the Overriding Safety Function

The historical information summarized in this section is based on the experience of the members of the ASME Task Force and on the seminal publication written by Mechanical Engineering Professor David Okrent from the University of California at Los Angeles, a long-time member of NRC's Advisory Committee on Reactor Safeguards (ACRS), in which served many of the AEC's early safety experts. [48] This historical recounting of the emergence of core cooling as the overriding safety function shows once again how the design basis of nuclear power plants has evolved over time to account for new information, new technology, and operating experience. Another useful reference for understanding the evolution in design features for providing defense-in-depth was developed at the Massachusetts Institute of Technology. It organizes safety systems according to reactor and containment design types and addresses the contributions to defense-in-depth by normal operating systems, engineered safety features, and special design features. It addresses operating light water reactors of western design located around the world. [49]

In the 1950s, when overseeing safety of the earliest nuclear power reactors, designers were particularly concerned with accidents involving uncontrolled reactivity excursions. Such events could cause a sudden large rise in reactor power, which could lead to fuel melting, structural damage of the reactor core, and breach of the reactor's primary system pressure boundary; the loss of the reactor coolant boundary could lead to core melting. There was also concern for core overheating and melting that might be caused by inadequate removal of heat associated with radioactive decay of fission products following reactor shutdown. Sabotage was also considered a possible cause of severe accidents in these early days, thus foreshadowing the protective measures added to power plants with the introduction of 10 CFR 73.55 in 1978 and the additional protective measures instituted after the events of September 11, 2001. [50]

The first power reactors were small enough that an exclusion distance of several miles precluded harm to people, should fuel melting occur by any means, though the reactors were housed in conventional buildings. However, as the size of reactors increased and the need grew to site them nearer to population centers where the demand for electricity existed, distance alone was not enough to ensure public safety. These circumstances led to introduction of containment buildings strong enough to hold in radioactive fission products, should a fuel melting accident occur. The first power reactor to be equipped with a strong containment building was a prototypical submarine propulsion reactor housed in a large steel sphere near West Milton, NY.

The first commercial nuclear power plant was the Shippingport Atomic Power Station in Pennsylvania. It was a pressurized water reactor (PWR) built by the Atomic Energy Commission and operated by the Duquesne Light Company. Containment vessels were included in the design of Shippingport and the first three privately-built nuclear power plants (Dresden 1, Indian Point 1, and Fermi 1). These plants had thermal power levels ranging from about 200 to less than 700 MW.

In 1956, Senator Bourke Hickenlooper of Pennsylvania questioned the AEC about the safety of the Shippingport plant. The Acting AEC Chairman, Dr. Willard F. Libby, responded, in part, "Under our regulations no license will be issued for the operation of any reactor, regardless of size and intended use, until the scientists and engineers who conceived and designed the reactor have made a complete evaluation of **all potential hazards** of their particular reactor, and of the adequacy of the steps they have taken in design and operating procedures to minimize the probability of occurrence of an accident which would result in the release of unsafe quantities of radioactive materials to the surroundings….**The financial incentive of the owners of the reactor to take all steps necessary to protect their investment, as well as to decrease their potential pubic liability, and the legal and moral responsibilities of the Commission** to protect the public from overexposure to radioactivity, are resulting in a system which is characterized by an attitude of caution and thoroughness of evaluation unique in industrial history." The Acting AEC Chairman then articulated three factors of the AEC's safety philosophy, i.e., recognize all possible accidents, reduce the probability of such accidents to an acceptable minimum, and, by a combination of containment and isolation, protect the public from the consequences of such an accident, should it occur. [51] The second of these three factors came to be known as accident prevention.

As the size of reactors continued to increase, an engineering design approach emerged to improve the capability of containments to adequately prevent fission products from escaping to the environment in the event of core melt accidents. The engineering changes included low-leakage containments, containment spray systems, and containment atmosphere cleanup systems, all of which were called engineered safeguards. The safety philosophy at that time was to prevent accidents and, if they nonetheless occurred, to contain and isolate fission products that could be released in a core-melt accident. These provisions were judged by the AEC to be sufficient to protect the health and safety of the public, even for complete melting of reactor cores of the size and on the sites then being proposed. That era soon ended.

In 1963, the AEC regulatory staff concluded that the 1347 MWt reactor proposed for construction as Unit 1 of the San Onofre nuclear power station in southern California could not tolerate a 100% meltdown of its fuel because the full release of the fission products in the core, when account was taken of the leakage rate of the reactor containment, would violate the AEC's newly-established siting criteria (10 CFR 100). Instead, said the staff, an emergency core cooling system had to be provided for the reactor to limit any core melting to 6%. Similarly, in 1964, a safety injection system was required to be added to the design of the 1473 MWt Connecticut Yankee power reactor to limit the amount of core melt to be considered in establishing compliance with 10 CFR 100. Soon thereafter, in response to a request from the AEC, the ACRS issued a November 18, 1964 letter, "Report on Engineered Safeguards." The letter addressed various ways of reducing containment pressures to ensure conformance with the siting regulations, but it was "ambivalent concerning core spray and safety injection systems," for cooling the core and limiting the amount of core melt to be considered in the siting evaluation. The ACRS letter expressed technical reservations regarding the efficacy of such core cooling systems. [52]

In July 1964, the ACRS reported on its review of a PWR nuclear power plant proposed for siting at Malibu, near Los Angeles. The ACRS letter said, in part, "The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area….The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system. Ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes the plant will be adequately protected." [53] The only mention of core cooling capability in this ACRS review was an emergency borated water injection system. The Malibu plant was eventually cancelled due to public opposition to its close proximity to Los Angeles.

Prior to 1965, the possible failure of the reactor pressure vessel was considered to be an incredible event and was not considered in safety analyses for nuclear power plants. There was no protection provided for such a failure, though it was expected to lead to core meltdown, loss of containment, and uncontrolled release of radioactivity. As commercial reactors continued to increase in size, this gap in safety assurance was widely recognized to be intolerable. In November 1965, the ACRS proposed to the AEC Chairman, Dr. Glenn T. Seaborg, three possible approaches to ensuring adequate protection for reactor pressure vessel failure: design containments to cope with missiles from pressure vessel failure; provide adequate core cooling or flooding that would function reliably in spite of pressure vessel failure; and, if containment breach could not be precluded, provide means of preventing uncontrolled release of large quantities of radioactivity to the environment. [54] After significant public debate, the AEC opted for a combination of quality control and failure protection to address the concern for reactor vessel failure. The quality control aspects were addressed by addition of sections to the ASME Boiler and Pressure Vessel Code to address nuclear pressure vessels and by heightened programs of inservice inspection of reactor pressure vessels. The failure protection aspects were addressed by criteria that emerged for so-called emergency core cooling systems (ECCS). These matters were resolved in the AEC's licensing of Unit 2 of the Indian Point nuclear power station located near New York City and Unit 3 of the Dresden nuclear power station located near Chicago.

In 1966, Indian Point Units 2 and 3 were the highest power PWRs to date (more than 3000 MWt), and they were to be located less than 30 miles from New York City. That same year, the AEC regulatory staff and Brookhaven National Laboratory (BNL) prepared WASH-740, a study of the consequences of severe accidents. One conclusion of the BNL work was that a core melt in a 3,200 MWt reactor would not only lead to melt through of the reactor pressure vessel but also to melt through of the concrete of the containment floor and on into the earth until enough material was involved to dissipate the heat. The conclusion that core melt and containment failure could be coupled led to a sea change in the safety philosophy for nuclear power plants. Added to the realization that a core meltdown in a high power reactor could penetrate the containment was the realization that metal-water reactions in the course of such a core meltdown could produce enough hydrogen to lead to an explosive challenge to containment integrity. The ensuing discussions among the regulatory staff, the ACRS, the Commissioners and the nuclear industry began to assume that some form of emergency core cooling system had to be included in the mix of engineered safeguards, to reduce the probability of core meltdowns for the large reactors then being reviewed for construction permits. At the time, the emphasis began to be placed on LOCAs as the most probable source of core meltdown. [55]

The concern that a core meltdown in a large reactor would likely lead to containment failure came to a head in mid-1966 when a member of the ACRS, during the Committee's review of Dresden Unit 3, recommended to his fellow members that either the power level of the reactor had to be reduced sufficiently so that a molten core would not penetrate the bottom of the containment vessel or an ECCS "far more reliable than any which now exists" had to be provided. The added conditions were to satisfy a requirement that "after a loss-of-coolant-accident at design power, there is as much assurance that core melt-down can be prevented as the reliability of conventional containment for relatively low-power reactors." In the end, both the Indian Point and Dresden units received construction permits, with provisos that new and more effective emergency core cooling systems be added to their design basis, before their owners returned with applications for operating licenses. As Dr. Okrent summarized the situation 25 years later, "…an approach was developed during the summer of 1966 which, in essence, created a major change in the engineered safety requirements for light water reactors and really set light water reactor safety on a new path. It had become important to make the probability of core melt much lower than it had previously been. The first two major steps, which were taken in connection with the Indian Point 2 and Dresden 3 reactors were: (1) to require much improved quality in the primary system, more inspection and much more leak detection in order to reduce the probability of a loss-of-coolant accident; and (2) to require a much improved emergency core cooling system in order to reduce the probability that a LOCA would lead to core melting. **This was the beginning of a continuing series of efforts that looked in ever-expanding directions for possible causes of initiating events that could lead to core melt, and sought measures to reduce the probability of such events."** [56] The new path to which Okrent referred was addition of ECCS to the design basis, i.e., it was better to prevent core melt (good design, good inspection, and good ECCS) than to mitigate core melt (containment). Thus, prevention continued to be the main strategy.

In October 1966, the AEC established a task force to address improvements of ECCS designs to prevent substantial meltdown and measures to cope with molten masses of fuel if the ECCS failed in its mission. The task force came to be called the Ergen Committee, after its leader William Ergen from Oak Ridge National Laboratory. The Ergen Committee concluded, among other things, "…that within the framework of existing types of systems, sufficient reliance can be placed on emergency core cooling following the loss-of-coolant, and additional steps can be taken to provide additional assurance that substantial meltdown is prevented." [56] To go along with improvements in ECCS design, the committee recommended improvements in assurances of primary system integrity (i.e., a continued emphasis on accident prevention). The committee was also the first to formally recommend studying filtered venting as a way to preserve containment function and long-term viability of core cooling. In ensuing years, the AEC's safety construct for light water reactors emphasized accident prevention through quality control of the reactor coolant system and high performance of ECCS. In 1974, the AEC completed a two-year rulemaking, including a lengthy public hearing, to establish acceptance criteria for ECCS performance and analysis methods for design basis LOCAs. [57] This safety paradigm, which emphasized LOCA avoidance, and preservation of core cooling if a LOCA occurred, endured until the core-melt accident at Three Mile Island in 1979.

4.3 Insights on Core Cooling from the 1975 Reactor Safety Study

The primary lessons learned from the Reactor Safety Study are listed in Chapter 3. They include the observation that the most important contributors to nuclear power risk are not the large LOCAs that underlie most safety system design requirements in the currently-operating plants. Rather, the dominant contributions to risk are combinations of more likely events, e.g., small LOCAs and loss of power, along with accidents involving coincident or cascading failures of multiple components or systems. Said another way, core cooling must be assured not only for large LOCAs, but also for a range of other initiating events that are more likely to occur and, if left unmitigated, can lead to core melt. For example, loss of offsite power combined with failure of emergency onsite power can ultimately lead to loss of capability to remove decay heat from the primary system, followed by increasing pressure and decreasing water inventory in the primary system, which can lead to core melting. A similar high pressure core-melt accident occurred at TMI-2, but for another reason—operator error—that defeated the core cooling function for a long-enough time that core melting occurred.

4.4 Insights on Core Cooling from the TMI-2 Accident

The information in this section is based on the experience of members of the ASME Task Force who participated in recovery from the accident and on the report of the NRC's Lessons Learned Task Force for the TMI-2 accident. [58]

The accident at TMI-2 demonstrated the fragility of several defenses against core melting, the most important being human performance. The accident started as a loss of feedwater flow. Operators soon learned that a maintenance error had left the steam inlet valve to the steam-driven auxiliary feedwater pump in the closed position rather than the open position required for its operation. Then, because there was no direct indication of its position available to the operators, they failed to observe that the pilot operated relief valve (PORV) at the top of the pressurizer had stuck open. High coolant flow out of the reactor through the PORV caused level swell in the pressurizer, which was indicated on level instrumentation in the control room. To protect the reactor coolant system from going water solid and over pressurizing, the operators turned off the ECCS that had actuated automatically upon loss of feedwater. With the PORV stuck open, water continued to escape from the reactor, and the main coolant pumps, which kept water flowing to the core for cooling, began to cavitate due to the large amount of steam that was accumulating in the primary system. Rather than reestablish ECCS flow to the reactor, to replenish the escaping water, the operators tripped the main coolant pumps. The loss of forced flow in the reactor vessel led to phase separation. By that time, the primary system had lost substantial water inventory, and the separated water level was not high enough to cover the reactor core, leading to core overheating and melting. This sequence of events showed the importance of human performance in ensuring reliable core cooling. Operators were handicapped in their ability to maintain core cooling during the accident, by inadequacies in their qualifications, training, technical support, emergency operating procedures, control-room design, and reactor instrumentation.

As the accident progressed, significant metal-water reaction in the overheated core liberated a large amount of hydrogen gas, some of which inhibited natural circulation in the primary coolant system and some of which found its way to the containment where a hydrogen burn occurred on the first day of the accident. Debris from the melting core accumulated in the reactor coolant system and the reactor coolant letdown system. Fortunately, it was not necessary to circulate the debris-laden-coolant outside of containment, which was required for use of the ECCS to cool the core, because forced circulation was reestablished in the primary system. This mode of cooling the core transferred heat to the secondary system and thence to the ultimate heat sink, while confining most of the core debris to the reactor coolant system. These circumstances demonstrate the difficulty of reestablishing core cooling during a core meltdown.

Eventually, nearly a half day into the accident, offsite managers of the utility and the NRC recognized the precarious cooling situation of the core and directed plant personnel to resume filling the reactor coolant system and to restart a main coolant pump to reestablish flow in the reactor coolant system. It took another 36 hours to realize from interpretation of damaged reactor instrumentation that restarting the main coolant pump had inundated and begun to cool the accumulated mass of molten core materials in the reactor vessel. The molten material was fully quenched about four days later. Natural circulation cooling of the core was reestablished about one month later. These circumstances demonstrated the capability to reestablish core cooling and interdict the course of core melting if operators are equipped with the knowledge and the means to do so.

The lessons learned from the accident at TMI-2 have enabled designers to improve the capability to ensure core and containment cooling. For example, the Generation III+ plants now beginning construction incorporate passive systems to provide core cooling for a minimum of 72 hours, without operator intervention, following design basis events, even in the case of a station blackout of indefinite duration. This critical function can be maintained indefinitely, with very simple operator actions from available equipment and sources of water. In a similar manner, containment integrity can be maintained indefinitely, being air-cooled from passive convective heat transfer. Designs also incorporate passive features to inundate the reactor cavity if the core overheats and melts, thus significantly reducing the failure probability of the reactor pressure vessel and protecting the containment liner and base mat in the unlikely event that the reactor pressure vessel fails.

4.5 Insights on Core Cooling from the Events of September 11, 2001

The events of September 11, 2001 led to generic and plant-specific analyses of the potential susceptibility of nuclear power plants to malicious, well-targeted, large explosions and fires. As described in Chapter 3, the nuclear industry took significant steps after September 11, 2001 to ensure continuation of core cooling in such events and has committed to strengthen these capabilities in the aftermath of Fukushima.

4.6 Insights on Core Cooling from the Fukushima Accident

The events at Fukushima show that the operator's highest priority should be to supply and maintain core cooling and that efforts to accomplish this, with any system or capability that can do the job, should be ongoing essentially indefinitely. That is, at any stage of an accident beyond the design basis, whether it be prevention of core melt, interdiction of core damage, or mitigation of core melt, the most important thing to do is to cool the core. Once the operators at Fukushima understood the tsunami had incapacitated the normal method of cooling the core, they tried various ways to "jury-rig" core cooling. Later, when they recognized that the cores in three of their units were severely damaged and recirculation cooling was impossible, they injected sea water and then clean water to cool the cores. Later still, they injected water to inundate and cool molten masses of core debris that had dropped onto the containment base mats. [15][16][17] [40] These circumstances demonstrated the need for additional equipment, to cool the core during the course of an accident beyond the design basis, such as alternative power and coolant delivery systems, clean water supply, contaminated water storage, access to an ultimate heat sink, and water cleanup systems. The accidents also showed the need for improved training, procedures, communication, and instrumentation, to understand the condition of core cooling and to aid its reestablishment and maintenance during the course of a severe accident.

4.7 Protection from Rare Yet Credible Events

One of the insights from the accident at Fukushima Dai-ichi was the inadequate protection of important plant safety systems from potential damage from a tsunami, including flooding of the electrical distribution system. Although less likely in other parts of the world, large tsunamis are an example of rare yet credible events that have the potential to present a severe challenge to nuclear power plant systems due to their cliff-edge effect. However, flooding from causes different than tsunamis also needs to be addressed.

Human nature is to judge the potential severity of rare yet credible events based on personal and readily available or widely known historical experience. For example, it is widely known that California and the West Coast are more prone to earthquakes than most other regions of the U.S. This experience does not, however, mean that severe earthquakes can be ruled out for plants in the eastern U.S., nor can tsunamis on the Atlantic coast or the Gulf of Mexico, simply based on observed experience. Likewise, a few hundred years of experience with storms that could cause extreme river flooding cannot be, and should not be, the limit of what is considered in nuclear power plant designs. However, lack of data makes judgments about what is sufficient a technical challenge.

By definition, few data points exist on the occurrence of rare yet credible events. Even probabilistic techniques, which have served the industry well in considering beyond-design-basis combinations of failures, have limitations due to lack of data for estimating the probability/consequence relationship for rare yet credible events, especially for rare natural phenomena. The uncertainties in estimating the probability of rare yet credible events can be very large, and traditional techniques that rely on mean values might not be sufficient to inform all design decisions, especially for events that present a cliff-edge challenge.

As described in Chapter 3, the margins inherent in nuclear power plant designs are an important element of the existing safety construct. The Fukushima Dai-ichi accident warrants a reconsideration of the margin inherent in current designs. Such reconsideration is ongoing, world-wide. In most cases, these margins provide some assurance that equipment and structures will function beyond their design requirements. This means, for example, that there might be significant margin for earthquakes and high winds. On the other hand, margins for large flooding events, from either internal or external sources, deserve further attention because, as shown at Fukushima, the potential consequence of catastrophic flooding can be extensive equipment failures.

It is the potential for common-cause failures, combined with uncertainties regarding event severity, that suggests special consideration be given to protective features for such events in the new safety construct. Further, depending on the uncertainties associated with these hazards, great care must be taken in using probability estimates as a means to establish designs, just as it might not always be sufficient to rely solely on a deterministic design basis. That is, other means of event interdiction could be judged to be needed to provide core cooling capability, if the design basis is exceeded.

4.8 Summary Comments

Accident prevention should continue to be the principal strategy in the New Nuclear Safety Construct. This will require that the design basis be thoroughly reexamined to ensure it includes adequate treatment of rare yet credible events and appropriate combinations of internal failures, including common-mode failures.

In addition, maintenance of core cooling should continue to be the overriding safety function in the new safety construct, i.e., to provide water to cool the core, from normal operations or shutdown all the way through to recovery from a severe accident. Thus, if water is pouring out of a break in the reactor coolant system, operators must keep new water coming into the system. If a core has melted and is challenging the integrity of the reactor pressure vessel or the containment, more water must be available to inundate and quench the melted core.

The Fukushima Dai-ichi accident has reinforced the longstanding principal safety approach of maintaining core cooling over a wide range of events, because this is the most effective method of preventing significant radioactive releases with their potentially-enormous socio-political and economic impact on society. The accident has indicated that the events now needing to be protected against include large fires and explosions, extreme natural phenomena, station blackouts of indefinite duration, and combinations of internal failures that can cause the loss of normal and backup core cooling that provide protection from the traditional design-basis events. This reasoning leads to the all-risk approach in the New Nuclear Safety Construct.

5 MANAGING THE UNEXPECTED – HUMAN PERFORMANCE

5.1 Introduction

Despite best efforts, there is a finite probability that a severe reactor accident, caused by either internal or external initiators, can happen, resulting in release of radioactivity to the environment. Such rare yet credible events are just as true for other human endeavors as they are for nuclear power. Many recent examples exist of the occurrence of highly improbable events with unforeseen loss of control, where human actions and decisions have contributed to, or ultimately led to, unacceptable consequences. Recent examples include the Deepwater Horizon fire, explosion, and oil leak in the Gulf of Mexico; inundation of New Orleans following Hurricane Katrina; crashes of the Space Shuttle Columbia and Concorde aircraft; and collapse of the World Trade Center buildings after terrorist actions on September 11, 2001. It is presently not possible to predict the occurrence of such events; furthermore, attempts to predict such events even if information is available will encounter significant uncertainty. The capability to predict and control an event becomes increasingly more difficult as the frequency of occurrence of the event decreases. Furthermore, people have low tolerance for involuntary risks, especially if the risks are due to failure of modern technologies, which are claimed to be provided with sufficient safety margins.

On the other hand, preplanned and well-executed human decisions constitute an important management feature to prevent and mitigate consequences of failure of engineered systems. Under crisis, decisions might be required to go beyond procedure-based approaches developed for normal operation and design basis transients and accidents. Reliance on pre-established procedures, controls and processes, coupled with rigorous training and reinforcement, can help reduce the possibility of error; but these precautions cannot completely preclude collateral events or errors, particularly when the events are far beyond those previously considered, e.g., extreme external initiating events.

Controlling the probabilities of human-based outcomes has been well described as part of a dynamic learning process. [59] Accumulation of knowledge and skill, both personal and organizational, is reflected in improved operations and lower risk exposure. Since errors cannot be entirely precluded, timely interdiction and front-line crisis management become necessary. These aspects of human performance set the framework for better understanding of what happened at Fukushima Dai-ichi in the aftermath of the Great East Japan Earthquake and Tsunami.

5.1.1 Traditional Approaches to Human Performance

There are many facets and methods of estimating and controlling human performance, including analysis of human factors, training, oversight, independent checks, symptom-oriented procedures, and probabilistic analysis. Recently, “human reliability” has been based on analyzing human errors on a task-by-task, item-by-item, and situation-by-situation basis, as commonly adopted for probabilistic risk analysis using event sequences. For these analyses, the probability of a correct or incorrect action is assigned at each significant step or branch point in the hypothesized evolution of an accident sequence. The probability of any action is represented and weighted or adjusted by situational multipliers representing stress, environment, and time pressures. It is practically impossible to describe the nuances, permutations, and possibilities behind individual and collective human decision-making, so the human-technological system must be treated as an integral system.

The fundamental point is that a focus on the human performance aspects of decision making must be included to reduce the propensity, rate, and opportunity for errors, and to significantly improve the safety and reliability of modern complex systems, such as nuclear power plants.

5.1.2 Responsibility, Accountability and Authority for Decision-Making in a Crisis

Clear and authoritative decision-making during rare yet credible events is critical to effective emergency management and risk mitigation. Decisions in a crisis must take into account both objective evidence and subjective risks.

In any decision-making environment, it is necessary to have clearly established accountability and authority. As a general rule, to ensure correct decisions are made, accountability during a crisis rises upward through a chain of command; whereas authority is delegated downward to allow proper actions to be taken by people with first-hand knowledge of the situation on the ground and closer to the crisis. The goal is to ensure that decisions are made by knowledgeable people based on the best available information and then implemented with skill, on a timely basis.

Some decisions, during and after nuclear accidents, have large socio-political and economic ramifications, including potential impacts from widespread radioactive contamination, evacuation of people, and land use restrictions. Other decisions involve plant safety, intentional releases of radioactivity to the environment to maximize plant resilience to further damage, and system abandonment or preservation (e.g., to vent containment, evacuate a control area, or distribute potassium iodide tablets). All of these decisions must take into account the wider implications mentioned above. Because any decision also takes into account, implicitly or explicitly, its financial and political consequences, external pressures also affect decision-making in a crisis. Thus, for example, confusion and differences of opinion by knowledgeable people or people in authority on the magnitude of an exclusion zone, on the need for or timing of a controlled release of radioactivity or an evacuation order can lead to mistrust and ineffectiveness during a crisis.

5.1.3 Organizational Human Performance

Four organizational levels are known to contribute to safe and reliable operation of nuclear power plants (as they do in other hazardous industries); each level is affected by human performance and decision-making. These levels could be considered nonphysical barriers against release of radioactivity to the environment. [60] The first level is workers and workgroups, the second is management/supervision, the third is independent internal assessment, and the fourth is external assessment. Figure 5 provides a depiction of organizational levels/barriers of defense-in-depth.

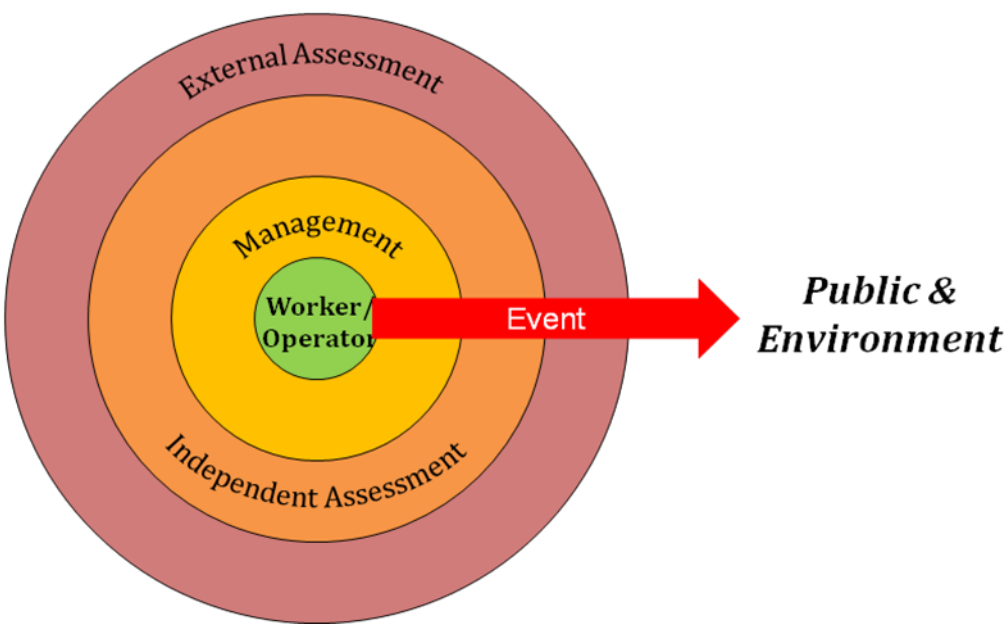

Figure 5 – Organizational Levels/Barriers of Defense-In-Depth

These organizational levels of defense-in-depth are complementary to the physical barriers of defense-in-depth (see Figure 6) that have been, and continue to be, an important part of the protection used in a nuclear plant to prevent release of radioactivity to the environment. These are the classic "hard" barriers that are easily understood and constitute key elements of the defense-in-depth strategy of the nuclear industry. These physical barriers of nuclear power plant defense-in-depth are the fuel cladding, the reactor coolant system pressure boundary, and the containment. As seen from both the TMI-2 and the Fukushima Dai-ichi reactor accident, these physical barriers can be bypassed or can fail, due to actions or breaches not fully appreciated beforehand. At Fukushima Dai-ichi, the failures included site location/elevation, sea wall height, flood protection for critical equipment, containment vent system design, and susceptibility to common-mode failure.

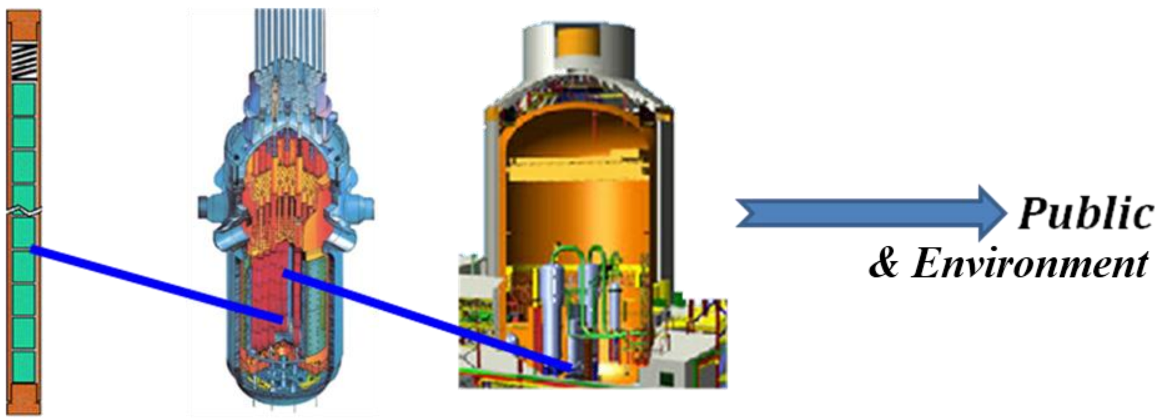

Figure 6 – Physical Barriers of Defense-In-Depth

In addition to the organizational and physical barriers of defense-in-depth, quality and safety are two other influential factors in establishing public confidence in a technology. Regulatory authorities have a Safety Culture Policy they work to engrain in organizations working in the nuclear industry; nuclear utilities have human performance programs that promote learning, self-reporting and corrective actions; in the U.S., INPO audits and accredits training programs and the NRC performs risk-informed baseline inspections to consider the safety significance associated with performance deficiencies as part of the Reactor Oversight Process (ROP). Collectively, these are important aspects of human performance in *preventing* accidents.

5.1.4 Organizational Failures

Typical indicators of organizational failures are well known, having been identified from major accidents in other industries, [61] and include the following:

1. Cost-cutting, failure to invest, and production pressures that impair process safety performance;
2. Executives who do not provide effective oversight of safety culture and major accident prevention programs;
3. Reliance on the low personal injury rate as a safety indicator that fails to provide a true picture of safety performance and the health of the safety culture;
4. Deficiencies in the mechanical integrity programs resulting in the "run to failure" of risk-significant equipment;

5. A "check the box" mentality, where personnel complete paperwork and check off on safety policy and procedural requirements, even when those requirements have not been met;
6. Lack of a reporting and learning culture, with staff not encouraged to report safety problems and fearing retaliation, resulting in failure to capture and act on lessons from incidents and near-misses;
7. Safety campaigns, goals, and rewards focused on improving safety metrics and worker behaviors rather than on process safety and management of safety-related systems;
8. Inadequate management response to the findings of surveys, studies, and audits; and
9. Ineffective management assessment of changes involving people, policies, or organizations that could affect process safety.

5.2 Human Performance in Nuclear Power Plant Accidents

Nuclear power plants normally are steady and uneventful machines to operate and manage. They produce vast amounts of electricity with few changes to their operation in the course of a typical day, and often for years on end. However, when the operational equilibrium is disturbed or a malfunction occurs, the entire organizational system is mobilized to maintain operations within predictable ranges. If this is not possible, the plant operators must be ready to take appropriate actions to correct the condition. The operators then have to diagnose their plant's status and the symptoms their plant is exhibiting, before they can decide on the proper sequence of actions needed to maintain the plant within a safety envelope. If needed, emergency management and support teams must identify and respond to gradations and ramifications of the event, and provide interdictory or mitigative actions, and possibly recommendations to other organizations, all with a focus on protecting the health and safety of the public.

Human performance expectations are significantly different for off-normal and accident conditions than for normal operation. In normal operation, command and control are less rigid. People can share responsibilities, communications are structured but not stressful, and accountability is routine. In abnormal operation, the command and control functions transition into a top-down structure. This transition is required because more deliberate team interactions and teamwork are needed to deal with the increased complexity of the situation. This structure is maintained for the operating staff when a further transition is made to emergency operations. As additional emergency response organizations and management layers are added, including offsite organizations, communications and interactions become more complex. These conditions lead to high dependency on the knowledge, skill, and judgment of the people on the scene to keep the "extended" accident management team informed of the status and progression of the emergency. In addition, as the accident progresses, the people on the scene must be able to implement the guidance provided by the top-down organizational structure quickly and adaptively. Experience has led to improved procedures, and better qualifications and better training in equipping people to deal with conditions of great stress, inadequate data, personal danger, time pressures, complex event evolution, and conflicting safety management attitudes and expectations. The transitions of responsibility from one organization to another are complex and difficult in a crisis environment.

Full-scope plant simulators have proven to be highly effective in training operators and emergency response personnel on response to transients and accidents. The simulators allow entire operating crews, along with plant management personnel, to respond to simulations of rapidly changing plant conditions. They test procedures, communications, and decision-making in a realistic environment. The simulators can also be used to challenge operating crews with unexpected combinations of equipment and instrumentation failures. Today, however, simulators have only limited capability to train people on severe accident progressions, such as core melt scenarios.

In summary, human decisions constitute important operational and management barriers to protect against failures of any complex engineering system in which physical barriers are the normal means used to protect workers and the public. Operational and management barriers that rely on processes, procedures, controls, and good practices provide a complementary defense-in-depth to the physical barriers.

5.3 Human Performance Lessons from Fukushima

The events at Fukushima Dai-ichi resulted in failure of the physical defense-in-depth barriers at three of the six reactors at the same time the organizational levels of defense-in-depth were severely challenged. [16] Human performance played a crucial role in the course of these events—at the power plant, in the neighboring region, and within governmental authorities in Japan and elsewhere—just as they did at TMI-2 and Chernobyl.

The initiating events resulted in extraordinarily-challenging physical conditions that affected human performance in many ways. As result, some human performance was heroic, both onsite and offsite. The operators and support staff worked in a darkened plant, navigating around falling debris during aftershocks, avoiding open manholes whose covers were blown off by the tsunami, detecting and bypassing high radiation levels, dealing with dramatic and dangerous hydrogen explosions, working in compartments flooded with radioactively contaminated water, all the while unable to inhabit their control rooms or communicate effectively with their management and associates. The event exceeded the bounds of the plant emergency procedures (as a result of loss of all AC and DC power, massive offsite and onsite infrastructure destruction, very high radiation, difficult environmental conditions, etc.), placing personnel in a situation where there was no pre-established guidance. They had no choice but to develop action plans and procedures in real-time, on the fly, as the situation evolved. The event also affected multiple units simultaneously and included interactions among units. Evacuations and protective action recommendations for offsite members of the public appear to have been conducted well, for the most part, under extraordinary conditions, without adequate lead time or reliable plant information. Management of the radiation exposure of workers onsite appears to have been handled well under the circumstances, despite damaged dosimeters, no computer database due to loss of power, and lack of radiation survey equipment and data.

Nevertheless, Japanese government reports indicate there were examples of less-than-acceptable human performance that contributed negatively to the course of events. [15] These human performance failures began well before the earthquake and tsunami occurred.

Inadequate human performance, starting more than 40 years before the events, was the principal cause of the accident at Fukushima Dai-ichi, that is, inadequate design for the size of tsunami that was experienced. The inadequacy of the original design for the tsunami and opportunities that were missed for correcting the situation in advance of the event will probably remain controversial for some time. [15][16][18] There were errors in site selection and preparation, plant design for the tsunami and ensuing consequences, accident analysis, regulation, and inadequate attention to international advice, well before the earthquake occurred.

While updated information existed that could have led to abatement of the principal causal factor (insufficient tsunami-wall height), it is not clear that this information would have prompted adequate near-term corrective action. In other words, while there was more recent information to indicate that a higher tsunami wall was needed, it cannot be positively stated that this would have led to increasing the tsunami wall to a height sufficient to mitigate the tsunami that was experienced.

Inadequate human performance after the earthquake and tsunami has also been recognized by Japanese safety experts [1] [15] [62][63][64][65] and others. In particular, at the national government level, it was learned that “the complicated structures and organizations can result in delay in urgent decision making.” [62] [64] The details of these lessons are documented in the references and are not repeated here; they are mentioned only to note that all nations can benefit from examining these hard-earned insights into human performance. In addition, cultural differences as they relate to organizational and operational performance can be important in understanding the underlying contributions to the severity of the Fukushima Dai-ichi accident.

Based on the extreme conditions at the Fukushima Dai-ichi site at the time, it would be difficult and perhaps unfair to judge human performance by site personnel during the accident evolution. Three reactor cores suffered severe meltdowns, yet radiological public health and safety were protected. However, extensive offsite contamination occurred because the operating organization was unable to interdict the course of events short of core melting or to prevent or mitigate the associated release of radioactivity off site. The operating staff was obviously hampered by very severe natural and "unexpected" phenomena. All of these, beginning with the core meltdowns, are considered unacceptable consequences for an operating nuclear plant anywhere, and they are the primary incentive for seeking a new nuclear safety construct.

Nevertheless, specific human performance deficiencies have been noted in reviews of the Fukushima events by competent authorities and likely will be the subject of improvements. These deficiencies include initial and subsequent engineering evaluations, plant and utility management, and governmental decisions:

- Inadequate technical review and safety management of historical tsunami data. The safety analysis prior to the Great East Japan Earthquake and Tsunami examined response only to earthquakes, not to earthquakes combined with other events, such as a tsunami [62][63];
- Failure to provide adequate preparation for severe accidents, considering all credible external events [62][63];
- Although the Severe Accident Management Guidelines (SAMGs) in Japan contained much of the guidance from the Boiling Water Reactor (BWR) Owner's Group, they failed to adequately address coping with interconnected events at multiple units, such as managing extended loss of offsite and onsite power, or responding to a loss of ultimate heat sink [62];
- Failure of multiple responsible organizations to adequately manage critical information and inform the public about long term management of land contamination, leading to a loss of public trust in authorities in local agricultural regions and major towns [15] [62];
- Inadequate management implementation of a Risk-Informed Safety Culture, a questioning safety attitude, and a combined deterministic and probabilistic basis for defense-in-depth [62][63] [65];
- Failure to implement adequate defenses for rare yet credible external events, for example, by giving inadequate attention to information provided by others on the need for significantly more capability to cope with large fires and explosions [62];
- Failure during the accident to limit offsite consequences, because of failure of containment venting systems and failure prior to the accident to adequately design components for those venting systems [62]; and
- Failure of multiple responsible organizations to properly estimate the evacuation requirements and their social, political, and radiological consequences [15]

The composite picture that emerges from this list is failure to achieve the proper organizational oversight by multiple responsible organizations, to adequately respond to rare yet credible events, i.e., a failure to acknowledge that rare events exceeding design basis can occur and should be prepared for in advance. In the nuclear power industry, this attitude could lead to inadequate preparation for controlling progression and mitigating consequences of severe accidents, not only for ensuring public health and safety, but also for protecting economic and political underpinnings of a society.

To reduce the propensity, rate, and opportunity for errors, and significantly improve the safety and reduce the risk of modern complex systems, there must be a focus on the human performance aspects of decision-making before, during, and after occurrence of rare yet credible events.

5.4 Summary Comments

The NRC Near-Term Task Force report [5], NEI's "The Way Forward" report [66], NEI's "FLEX" conceptual response to lessons learned from the Fukushima Dai-ichi accident, and the new INPO initiative on Emergency Management Excellence address a number of items related to human performance, pertaining to the existing requirement to provide adequate protection of public health and safety. The details of these items can be seen in the referenced reports.

The actions, decisions, and effectiveness of ensuring adequate core cooling and preventing release of large amounts of radioactivity must be emphasized as the highest priority in supporting the new safety construct. Knowing what is now known as a result of the Fukushima Dai-ichi accident, effective human performance requires decisions and actions that implement both physical and operational capabilities to restore in-plant electrical distribution systems and to provide cooling to the core and other essential equipment for as long as is needed to prevent a major disruption of society from large radioactive releases.

6 MANAGING ALL RISKS

6.1 Introduction

Accident management, in its broadest sense, refers to actions taken in response to an initiating event that causes an upset condition in a plant, including coordination of systems, equipment, and personnel. Since the accident at TMI-2 in 1979, the nuclear industry has placed a great deal of emphasis on accident management to first prevent and then mitigate accidents that could lead to the escape of radioactive materials into the environment, and thereby to protect the health and safety of the public. As discussed in Chapter 3, accident management includes interdiction of an accident sequence in progress to limit its consequences. To manage accidents and keep fission products within the plant, operators are trained and equipped to preserve the fuel cladding, the reactor coolant system pressure boundary, and the containment. Accident management also includes mitigation of releases, of radioactivity to the environment. Thus, accident management provides layers of protection to reinforce the defense-in-depth approach used in design of nuclear power plants.

6.2 Defining Accident Management

Accident management for nuclear power plants involves at least three phases in the progression of an accident, as shown in Figure 7.

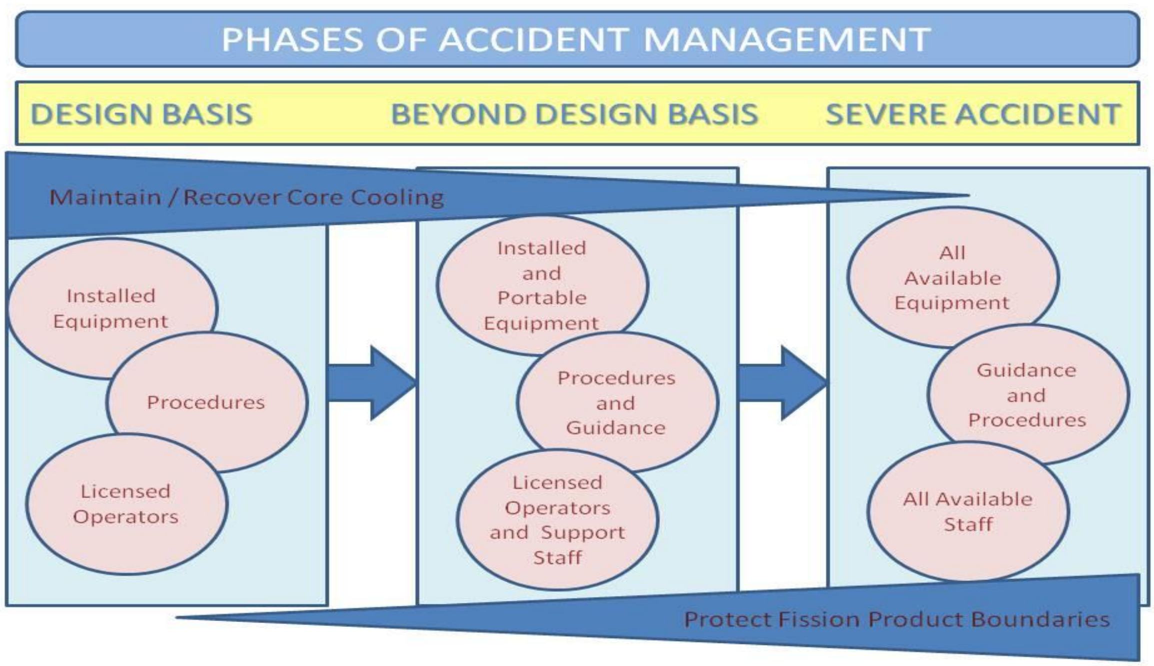

Figure 7 – Phases of Accident Management

As shown in the first phase, accident management addresses events within the design basis used to design, construct, and operate the plant. The second phase includes a range of events beyond the design basis, as those terms are defined in Chapter 3. The third phase includes severe accidents that involve core melting. Accident management includes the equipment and systems needed to mitigate an event plus the instrumentation and personnel to diagnose the event and activate the equipment to perform the required safety functions. To support accident management capabilities, equipment must be properly maintained, to ensure it is available when needed, and people must be properly trained to ensure they can use their procedures and guidance.

Accident management is formalized through procedures and guidance for plant operating and engineering staff to follow in off-normal or accident conditions. Accident management procedures and guidance consist of Abnormal Operating Procedures (AOPs) to address response to issues with single systems, Emergency Operating Procedures (EOPs) to address maintaining core cooling and subsequent recovery from events that cause a reactor trip or safety system actuation, and SAMGs to address protection of plant fission product boundaries in the event of a core damage accident.

The accident management procedures and guidance are typically symptom-based. That is, the plant operating staff is directed to respond to plant behavior observed through instrumentation rather than reconstructing the steps that brought them to that behavior. Therefore, accident management is equally applicable to externally-initiated events (e.g., earthquakes, external floods, and high winds) and to internally-initiated events (e.g., pipe breaks, internal flooding, and faulty operation of control rods).

6.3 History of Accident Management

The need for effective accident management was realized following the accident at TMI-2 in March 1979. During the initial stages of the accident, there was confusion among plant operators, plant engineering staff, and offsite authorities, regarding the status of the plant and actions needed to bring the situation under control. Although precursor events had occurred at a similar plant, and analyses had been performed that predicted the course of the event, this information had not been shared with the plant staff to enable the operators to effectively diagnose the situation and take appropriate actions to bring the plant under control before core damage occurred. Because the desired response of the operators was not proceduralized, they responded to the symptoms based on an incorrect determination of the risks that existed at the time. Acting on an increasing pressurizer level indication, the operators slowed and then stopped adding water to the core, i.e., they shut off the ECCS. Eventually, loss of water from the primary system through a stuck open relief valve led to inadequate core cooling and melting of some of the fuel in the core. The operators were more concerned with preventing the reactor coolant system from overfilling and losing pressure control than with maintaining core cooling.

Following the TMI-2 accident, significant advances were made in accident management, including development of symptom-based EOPs to maintain core cooling and containment integrity, for design-basis and beyond-design-basis events. The advances also included development of SAMGs to maintain fission product barriers in the event of core damage, and improvement of offsite emergency plans to protect the health and safety of the public in the event of a release of radioactive materials from the plant. To complement the new accident management tools, changes in plant systems and components were also made to assist the plant staff in diagnosing and responding to accidents. Finally, since the TMI-2 accident, there have been significant efforts to examine accident precursors to determine if accident management capabilities can be further improved. The importance of establishing and maintaining containment integrity, which is the final fission product boundary, became one of the cornerstones of accident management after the TMI-2 accident.

As a result of the socio-political and economic impact of the uncontrolled release of radioactive material from the Chernobyl accident in 1986, the importance of the containment to provide a final barrier was clear to all. The Chernobyl accident led to significant improvements in accident management capabilities in Europe, to reduce the probability of major releases of radioactivity to the environment. These improvements included filtered containment vents and hydrogen-control equipment in many plants. In addition, special backup systems were installed in some countries to maintain core cooling in the event of a beyond-design-basis external event or terrorist activity, by hardening the structures containing these diverse systems against extreme events. The U.S. plants were subjected to fewer changes after Chernobyl than the European plants, because U.S. utilities had already implemented significant changes after TMI-2 and believed their containment structures were robust and would protect against severe accident consequences.

Completion of high-quality, plant-specific PRAs in the early 2000s also contributed to advancements in accident management. Plant-specific vulnerabilities or high-risk situations were identified and remedied, either through design changes or by creation of guidance and procedures. The Configuration Risk Management Programs (CRMPs) were another important contribution to accident management. The CRMPs were put in place at all U.S. plants to help ensure that high-risk configurations were identified and addressed during maintenance and testing of plant safety systems. The CRMPs help to ensure that a proper level of defense-in-depth for mitigation equipment is readily available to maintain core cooling and containment integrity in the event of an accident initiated during any mode of plant operation.

Additional advances in accident management capabilities in the U.S. were made following the terrorist attacks on the World Trade Center in September 2001. The capability to maintain control of the plant following an event that damaged installed equipment and internal electrical infrastructure over a broad area of the plant was implemented through use of portable pumps and generators that could be readily connected to major plant systems. Extensive Damage Mitigation Guidelines (EDMGs) were created to provide the operating staff with proper steps for using this new equipment. Thus, where they are implemented, EDMGs have become part of accident management resources. However, this increased accident-management capability was mostly limited to the U.S.

The foregoing review of accident management suggests that the practice has been subjected to a systematic and continuing improvement process, learning from good and bad experiences. The current accident management capabilities, in terms of equipment, procedures, and guidance, and personnel training and qualification, have been aimed at addressing lessons learned from past events. Based on the accident that occurred at the Fukushima Dai-ichi plant, there is an apparent need to further improve accident management with capabilities to cope with rare yet credible events not yet addressed.

6.4 Accident Management at Fukushima

There were two key accident management vulnerabilities at Fukushima that contributed to the severity of the accident. First, the possibility had not been considered that the plant could be completely cut off from offsite assistance in the first hours after the event. Second, the possibility had not been considered that the plant would suffer the severity of damage caused by the tsunami. [15][16][17][18][62] [64][65] More precisely, there were no effective plans or equipment to handle such severe damage to the infrastructure and safety systems of six units at the site, simultaneously. In addition, there was no effective plan to provide assistance from offsite, in a timely manner, under the disastrous conditions caused by the combination of earthquake and tsunami. [15] These factors are examined in more detail below.

6.4.1 Multi-Faceted Disaster

The emergency infrastructure of Japan was not prepared for an event that crippled both the multi-unit station and the surrounding areas. Therefore, when the plant features to be used for accident management were damaged by the tsunami, there were limited capabilities for immediate assistance from offsite. In addition, onsite and offsite areas were strewn with debris, preventing accident management capabilities from being rapidly implemented, even after they became available. Portable equipment that could have been used to mitigate the consequences of the accident was either destroyed by the tsunami or the hydrogen explosions, or was prevented from being used in a timely manner by a combination of factors, including delayed decision-making. [63][64][65]

6.4.2 Accident Measures Against a Severe Tsunami

Fukushima Dai-ichi had insufficient defenses or offsite supporting capabilities to withstand the severe effects from the Great East Japan Earthquake and Tsunami. Even though historical records contained evidence of tsunamis of this magnitude, and of tsunamis exceeding the five-meter sea wall, comprehensive protection against such tsunamis was not provided. [64][65] Nor was there adequate flood protection of key equipment needed to cool the cores in the four units located nearest the ocean. Furthermore, no systematic, procedure-based coping strategy existed for dealing with consequences of such an event. For example, portable pumping systems (fire trucks) were available on site, and provisions had been made to connect the systems to the plant, but there was no established procedure for accomplishing these actions. The owner of these units, as well as other nuclear plant owners, did not prepare for the possibility that the safety-related electrical distribution system, and many of the plant safety systems, could be rendered inoperable by a single event. Also, the possibility of losing all AC power for an extended period of time, and the resulting depletion of the plant batteries (i.e., all DC power), was not considered in preparations for accident management.

In addition to the lack of severe-accident measures to cope with an extreme event of this magnitude, the authority and responsibility to implement some of the accident management measures were not well defined. Also, the process for considering input from the federal government was not well defined and understood.

6.4.3 Plant Conditions

The design of mitigation equipment did not include contingencies for a complete loss of support needed for implementation of accident management strategies. The prolonged loss of AC power, combined with the initial impact of the tsunami, led to a loss of DC power, which resulted in the loss of control room indication of plant conditions. The operators also were unable to operate equipment remotely. Moreover, some of the valves required for operation of the cooling systems and the drywell venting path were solenoid valves not easily operated in a manual mode with loss of control power. The design also did not consider that these valves might have to be locally-operated in a post-core-damage radiation environment; the high-temperature and high-radiation environment resulted in significant delays in operation of these solenoid valves and radiation exposures to plant workers.

6.5 Lessons in Accident Management from Fukushima

Prior to Fukushima, risk studies typically had shown the probability of an accident that would result in a major release of radioactivity was sufficiently low that additional consideration of accident management measures was not required. In the U.S., the “standard” for accident management improvements became the analysis of Severe Accident Mitigation Alternatives (SAMAs) required by the NRC as a condition of license renewal. Site-specific cost-benefit thresholds were used that limited implementation of SAMAs. More importantly, licensees could defer making improvements that did not directly relate to aging, because aging is the main focus of license-renewal determinations. These limitations in the U.S. have deterred significant hardware changes and enhancements in severe accident management.

As the Fukushima Dai-ichi accident showed, there were weaknesses in that approach. Because the PRAs used for SAMA analyses were limited in scope, events beyond the design basis caused by extreme external initiators were not analyzed with the same degree of rigor used for events caused by internal initiators. Scoping analyses led to the conclusion that a beyond-design-basis external event that would overwhelm the accident management capability was extremely unlikely, and that additional coping capability was not needed to provide adequate protection of public health and safety. These scoping analyses did not include all external hazards and did not adequately address some extreme external hazards. At the time, the consequences of these extreme external hazards were underestimated. [2] [15][16] [64] Thus, layers of protection against unacceptable consequences were not implemented for a range of rare yet credible external events. As a result, the defense-in-depth against these beyond-design-basis external hazards would appear to be inadequate to prevent severe core degradation. This consideration supports the ASME Task Force proposal that an all-risk approach be used.

Reactor operators and other plant staff typically receive training and perform exercises for plant conditions beyond the scope of their EOPs. Such training and exercises vary in technical content and frequency. There was confusion among the operating staff at Fukushima Dai-ichi on the transition from their EOPs to other procedures and guidance, such as SAMGs, when, due to lack of instrumentation, the required symptoms were not discernable. [15] [17] [63] As discussed below, in the post-Fukushima global response, there is concern that the existing strategies involving SAMGs and EDMGs (implemented in the U.S. after September 11, 2001) are not sufficiently effective for dealing with an extreme event such as experienced at Fukushima.

The operators at Fukushima Dai-ichi were also handicapped in using the available accident management guidance, because current EOPs and SAMGs in Japan and elsewhere assume that some AC power is available or can be quickly recovered and that DC power is always available for continued operation of instrumentation. [67] Without instrumentation to provide information regarding key plant parameters, the EOPs and SAMGs become more difficult to use and could even break down. The EOPs for most U.S. plants are developed around AC power recovery in accordance with the NRC's Station Blackout Rule (10 CFR 50.63), which has a relatively-short coping time (e.g., up to 8 hours). The EOPs have not been developed for the case in which no AC power is available, and do not address an extended station blackout.

The inadequacies of accident management guidance, equipment, and instrumentation distracted Fukushima Dai-ichi plant staff whose primary problem was to respond to inadequate core cooling. For example, plant staff had inadequate user training on SAMG equipment and inadequate control room simulator training on severe accidents. They also had to deal with spent fuel pool cooling and potential damage to the spent fuel pools. These and other extraneous matters reduced the staff's focus on maintaining core cooling, the overriding safety function for nuclear reactors. [2]

6.6 Post-Fukushima Global Response

Following the Fukushima Dai-ichi accident, utilities and regulators world-wide began to undertake an examination of measures to address accident management for a long-term disabled electrical power distribution system. The objective of these assessments was to determine the readiness of the plants for beyond-design-basis events, as discussed in INPO IER 11-1 self-assessment guidance and the NRC TI 2515/183 and TI 2515/184 inspection process. [68] [69]

In general, it was found that accident management capabilities (equipment and procedures) were in place that could have been important in either protecting the core or protecting the containment, and thereby could have prevented the uncontrolled release of radiation to the environment in conditions such as those experienced at Fukushima Dai-ichi. In particular, accident management capabilities relying on installed equipment and EOPs, at least in the U.S., have been maintained in a high state of readiness. However, additional accident management capabilities to deal with core damage accidents or extreme events that could damage key installed equipment have not been maintained at the same high level. For example, post-Fukushima assessments conducted at U.S. facilities showed that, at some plants, portable pumps were available as intended but hoses or connections for attaching the pumps to plant systems were missing. In other cases, the plant staff was not properly trained to conduct the accident management activities so that the likelihood of successful implementation could be questioned. In Western Europe, the results of the stress tests that have been completed to assess the capability of each plant to withstand extreme external events reveal that there are significant variations in accident management capabilities amongst the participating plants. [70]

Stress tests were undertaken in many countries after the Fukushima Dai-ichi accident to determine the safety margin above the plant design basis for external events. In other countries, examinations were performed of the safety margin for a loss of all AC power that extends past the design basis coping time. These examinations revealed that, when accident management strategies are credited, significant margins above the plant design basis exist. These examinations also revealed vulnerabilities for extreme external events and provided opportunities to improve accident-management capabilities. The stress tests have also revealed that there are levels of extreme events where current accident management strategies are not effective, due to either lack of operable equipment or inability of plant operating staff to diagnose and implement the strategies. These are issues that could be improved by using an all-risk approach. [70]

In the U.S., the nuclear industry has developed a response structure to identify and apply lessons learned from Fukushima, through self-examination. “The Way Forward” [66] describes the industry approach, called FLEX, to enhance accident management capabilities based on the lessons learned from Fukushima. [71] The FLEX approach is to provide a diverse and flexible accident-response capability that would provide a backup to permanently-installed plant equipment that might be unavailable following extreme natural phenomena (e.g., earthquakes, flooding, high winds) and would supplement the equipment already available for responding to malevolent acts. [72] The FLEX approach would include design measures to provide multiple means of obtaining power and water needed to fulfill the key safety functions of maintaining core cooling, containment integrity, and spent-fuel-pool cooling.

The FLEX approach involves development of new accident management techniques that improve the capability of a plant to survive an extended loss of all AC power as a result of extreme external events. It includes equipment to respond to the challenges; procedures and guidance; equipment readiness, storage, and transportation; and training. The increased equipment capability will consist of installed equipment, portable equipment stored onsite, and portable equipment in regional and national centers. This approach recognizes the need to provide accident management capabilities when the onsite and the offsite infrastructure are severely damaged. The FLEX approach is phased, to consider the immediate need to maintain core cooling and containment integrity and the potential need to maintain these capabilities for an extended period.

The FLEX approach addresses the possibility that an event could be more severe than known historical events. In the U.S., plants have been challenged in recent years by severe hurricanes (e.g., Turkey Point in 1992), floods (e.g., Ft. Calhoun in 2011), tornadoes (e.g., Browns Ferry in 2011) and earthquakes (e.g., North Anna in 2011). While core cooling and containment integrity were not challenged by these events, it is possible that worse events could challenge safety. Thus, it is important that the FLEX approach be prepared for rare yet credible events that could lead to severe consequences. The FLEX approach is on the right track by being less concerned about probability of occurrence and more concerned about providing layers of protection against rare yet credible events, enabling plants to prevent loss of core cooling, preserve containment integrity and mitigate consequences, should protections against design-basis events fail.

6.7 Beyond the Present Response

The lessons learned from the Fukushima Dai-ichi accident point to the need to consider rare yet credible events that can cause infrastructure damage to the plant and the surrounding communities that might be depended on for support. However, the key accident management lesson that should be taken from this accident is the need to prevent large radioactive releases that could cause major disruption of society with attendant socio-political and economic consequences of unacceptable proportions. The only reasonable manner in which to approach this issue is to use an all-risk approach. Because socio-political and economic impacts cannot be confined by geo-political borders, this approach should be applied on a global basis.

The proposed all-risk approach to accident management, with appropriate consideration of probability of occurrence, associated uncertainties, and potential consequences, including cliff edge effects, would address a broad range of challenges to safety of nuclear power reactors and spent fuel facilities, including internal hazards, external hazards, and security threats, during all modes of plant operation. These challenges would be addressed in a risk-informed manner for both design-basis events and events exceeding the design basis, including rare yet credible events. The effectiveness of the capability to mitigate challenges and their consequences for all risks is key to identifying the appropriate enhancements to be considered. This approach is likely to result in changes to all phases of accident management, as identified earlier in Figure 7, including equipment, procedures, guidance, and training and qualification of personnel. Finally, there is a need for a uniform global standard of excellence for accident management capability including the definition of the level of extreme external events against which plants, plant workers, and the general public must be protected. That is, there should be accident management measures in place, and maintained in a state of readiness, as part of the design basis and coping capabilities to deal with rare yet credible events.

6.8 Summary Comments

The Fukushima Dai-ichi accident has revealed that current accident-management capabilities have weaknesses, and should be improved. The world-wide nuclear industry and national regulators are working to address these weaknesses. Strengthening accident management capabilities should, in most cases, result in a more robust capability to deal with design basis events as well as beyond-design-basis events. Improvements in accident management capabilities are being studied on a country-by-country basis (or in some cases a regional basis) and will undoubtedly result in different levels of implementation world-wide.

The future of nuclear power could depend on any one plant’s ability to cope with a beyond-design-basis event. Therefore, there is a strong need for global unity and excellence in defining and maintaining a high level of accident-management capability.

7 EMERGENCY PREPAREDNESS

7.1 Introduction

Emergency preparedness (EP) is an essential part of the overall defense-in-depth philosophy applied to design, construction, and operation of nuclear power plants. The U.S. NRC's requirements for EP are contained in 10 CFR 50.47 [73] and 10 CFR 50, Appendix E [74]. The requirements apply to the plant licensee and, in addition to specifying emergency planning zone (EPZ) size, contain a set of 16 planning elements that cover a range of onsite functions, including organization, equipment, training, and interfacing with offsite agencies. Requirements for the offsite agencies in the U.S. are the purview of the Federal Emergency Management Agency (FEMA) and are contained in FEMA regulations.

The EP planning elements are designed to provide protective measures for public health and safety to be taken in the event of a radiological emergency. EP is discussed herein for several reasons: (1) the new safety construct is intended to enable a safety outcome in an all-risk approach that has ramifications for EP; (2) in addition to elevating awareness of socio-political and economic effects of severe accidents, the new safety construct includes the need to protect against radiological public health effects; and (3) there are significant commonalities in EP functions and accident management functions, not the least of which is that the onsite emergency response organization (ERO) oversees both emergency response and accident management in the event of a radiological emergency.

The ASME Task Force views on EP have a global perspective, but with emphasis on the situation in the U.S.[3] This perspective includes a listing of lessons learned from the Fukushima Dai-ichi accident and a discussion of a number of EP areas that encompass most of the lessons learned and the differences between current practices and what the ASME Task Force recommends for consideration in the new safety construct.

Following the TMI-2 accident, considerable attention was focused on establishing a robust EP program with a formal Emergency Plan for each nuclear plant site and for states and local government agencies within the EPZ. Further changes evolved in the aftermath of the September 11, 2001 terrorist attacks in the U.S. As a result of these efforts, EP has become a mature, effective program. Now, with the experience of the Fukushima Dai-ichi accident, it is appropriate to again consider how EP can be improved. Any initiative to improve EP—indeed potential plant improvements of any kind—should be subjected to an objective, structured, and integrated review, to ensure that the proposed initiative will provide the intended benefits and will not adversely affect some other element of defense-in-depth. There should be a clear and quantifiable safety benefit for any such initiative before it is implemented. In addition, the overall costs of the initiative must be weighed against the safety benefit that will be obtained. Broad-based stakeholder engagement will be needed on EP matters, to ensure that the improvements implemented are cost effective and meet the objectives of improved safety.

7.2 EP-Related Lessons from Fukushima Dai-ichi

There are a number of important EP-related lessons to be learned from the Fukushima Dai-ichi accident. Table 4 lists those that the ASME Task Force considers to be most important, along with the relevant NUREG-0654 (Criteria for Preparation and Evaluation of Radiological Emergency Response

[3] Emergency planning is generally a requirement for nuclear plants worldwide. The specifics of EPZ size and EP requirements, principles, and practices vary from country to country and are not discussed herein. The IAEA has published a number of documents [76] that contain guidance and advice on development of an emergency response capability, based on potential nature and magnitude of the risk.

Plans and Preparedness in Support of Nuclear Power Plants) planning standards, where applicable. Many of these lessons have been recognized by others, such as the NRC [75], the NEI [71], and international groups, including the IAEA. [76] While priorities are still being discussed among stakeholders, a number of these lessons are already being addressed. Others—in particular the bottom five, which are the shaded rows in Table 4—are perhaps less recognized and are not addressed in NUREG-0654, but are nonetheless important, in the opinion of the ASME Task Force. Section 7.3 provides the ASME Task Force's views on a number of these lessons, which are considered to be of particular importance to the new safety construct proposed in this report.

Table 4 – List of Important Fukushima EP-Related Lessons

Fukushima Lesson	Relevant NUREG-0654 Planning Standard
Staffing for multi-unit events including extreme external events that could disrupt local infrastructure	B
Protective measures and equipment for emergency responders during multi-unit events	H, K
Command and control for multi-unit events	A, B
Dose assessment capability for multi-unit events	I
Need for improved onsite and offsite radiation monitoring including ac independence and real time availability (via internet or satellite)	I
Communications equipment effectiveness during a prolonged Station Blackout (SBO)	F, H
Need for accurate, automated, real time data on plant status	Not addressed
Drills and training under more realistic accident conditions	N, O
Adequacy of EP facilities during prolonged SBO and multi-unit events	H
Need for enhanced emergency response resources in light of potential for disruptions of onsite and offsite infrastructure	C
Need for enhanced EP decision-making framework including expansion of response beyond plume exposure EPZ and recovery and reentry	A, B
Gaps in public awareness of radiation and radiation safety	G
Need for better scientific basis for reentry (return home) criterion including low level radiation effects	Not addressed
Need for improved crisis communication systems recognizing the revolution in social media of the last decade	Not addressed
Need for building transparency and public trust in nuclear safety	Not addressed
Need for updated basis for EPZ size	Not addressed
Need for risk-informed, performance-based approach to EP	Not addressed

As noted in the introduction above, any plant or EP-process-related changes evolving from the Fukushima lessons learned must be assessed in the aggregate. Such an assessment would determine whether, and to what extent, improvements should be implemented on the basis of increasing protection of public health and safety and, beyond this, whether additional improvements should be implemented based on socio-political and economic impacts, which are a central aspect of the new safety construct. A key part of such an assessment is engaging stakeholders to ensure appropriate consideration of costs and benefits of the actions.

7.3 EP-Related Lessons in the New Safety Construct

This section discusses six EP areas that encompass most of the EP-related lessons in Table 4 and the differences between current EP practice and what the ASME Task Force recommends for the new safety construct. These six areas are: infrastructure, drills and training, long-term habitability, crisis communications, updated basis for EPZ size, and development of a risk-informed, performance-based approach to EP. Building public trust is addressed in Chapter 8 of this report, along with other cross-cutting issues. All of the Table 4 lessons learned should be considered by the global nuclear power community, with the ones discussed below and in Chapter 8 being of particular importance to the new safety construct.

7.3.1 Infrastructure Improvements

Among the highest priority areas for EP improvement in the new safety construct is the infrastructure (equipment, facilities, services, information technology systems) that is necessary to facilitate and allow emergency response functions to be carried out in the event of a severe reactor accident initiated by rare yet credible events derived from an all-risk approach.

Even with the benefit of decades of operating experience and insights from PRA, it is difficult for the nuclear community (as for any other complex systems technology) to anticipate all possible events or combinations of events that could cause a challenge to safety at a nuclear plant or possibly lead to an accident. Because of this, there is a need for a diverse, flexible, emergency response program that takes advantage of modern technology and for which there is high assurance that the system will exist and be functional under extreme, unexpected conditions that could cause the accident in the first place.[4] In addition to security events, this flexibility should encompass conditions brought about by low-probability, high-consequence external events, such as extended station blackout (SBO), including loss of all AC and DC power, a damaged site environment and perimeter, and no land-based site access for a substantial period of time.

Examples of potential EP infrastructure improvements that provide enhanced ability of the program to provide its required functions under unexpected conditions are as follows:

- The site's EP program should be self-sufficient for a range of severe accident conditions (e.g., SBO, multi-unit events, damage from external events) for an extended period of time (for as long as necessary), long enough that there is high assurance that necessary assistance can be supplied from offsite. This would include assuring that emergency response personnel can be protected from radiological exposure during this time; that there is sufficient access to water, food, and sanitary and sleeping facilities; and that emergency response equipment and facilities can function under the extreme conditions.
- There should be confirmation of the effectiveness of offsite protective actions in the EPZ under conditions of widespread, extended blackout conditions.

[4] A crucial aspect, perhaps the most important part, of a diverse, flexible emergency response program is maintaining key safety functions (core cooling, containment integrity, spent fuel pool cooling). These key functions must be coordinated and integrated with onsite and offsite EP. Maintaining key safety functions as part of emergency response is addressed in Chapter 6 of this report.

- The onsite and offsite communications infrastructure should be able to function under a range of accident and site conditions for an extended period. Examples of communications infrastructure capabilities that should be considered are discussed in Chapter 8.
- Equipment necessary to perform EP functions should be powered from reliable electrical sources with the capability to operate under a range of accident and site conditions such as prolonged SBO. This might include instrumentation used to monitor radiation levels on and around the site, instrumentation for monitoring meteorological conditions, computer equipment and software programs used for performing plume analysis and dose predictions, and equipment to assure automated transmission of accurate, real-time data from the site to offsite decision makers.
- Consideration should be given to a modernized network for radiation monitoring onsite and at the site perimeter (e.g., gamma chamber type detectors with an emergency powered Wi-Fi capability), to assure that the radiation monitoring capability is maintained under extended SBO conditions, to facilitate the potential need to expand monitoring to a wider area, and to allow more immediate communication of monitoring results to decision makers and stakeholders.
- Regional asset support centers should be considered that could be mobilized as necessary to provide assistance from offsite within the agreed upon coping time, e.g., 48 to 72 hours.
- A review of EP command-and-control and decision-making authority in emergencies should be performed. Specifics in this regard are provided in Section 8.3.1.

7.3.2 More Realistic Drills and Training

Effective implementation of the onsite and offsite EP program requires those with critical roles in executing the program to be well-prepared to perform their roles under emergency conditions. Onsite ERO personnel, as well as offsite responders and local, state, and federal officials must receive in-depth training on their responsibilities, and have the opportunity to practice execution of these responsibilities in drills and exercises that simulate real accident conditions to the extent feasible without impacting plant operation.

Since the TMI-2 accident, the U.S. nuclear industry has implemented a challenging program of training, drills, and exercises to ensure that the site ERO and offsite agencies are well-prepared for emergencies. Over the last several years, the drill and exercise program has been undergoing revision to address postulated losses of large areas due to fires and explosions. In light of the Fukushima Dai-ichi accident, the industry should critically reexamine existing training, drills, and exercises in three key areas that could warrant further improvements: (1) events that challenge the integrity of multiple units at a site; (2) events that unfold over a more protracted period than has traditionally been considered in drills and exercises; and (3) events such as severe external events and SBO, that result in significant disruptions of site access or infrastructure that support response to the event.

Some specific examples of the types of changes in training, drills, and exercises that should be considered, to address the three key areas, are as follows:

- Drills should consider accident scenarios in which there could be significant disruptions to onsite and offsite infrastructure and challenges to multiple units, such as extreme external events and station blackout.
- Drills should address slower-developing accident scenarios with radioactivity releases that challenge onsite and offsite emergency response over a prolonged period. The experience of Fukushima and recent severe-accident analytical studies, such as NRC's State-of-the-Art

Reactor Consequence Analysis (SOARCA), [77] have shown that accidents develop more slowly than typically considered in drills and exercises.

- Given that U.S. industry is currently contemplating addition of regional centers to provide supporting equipment and resources, the drill and exercise programs should incorporate changes to include demonstrating deployment of such equipment, including preparations to connect the equipment onsite under potentially adverse conditions, such as high ambient radiation levels and presence of extensive debris and other disruptions that could limit access to the areas in which equipment connections are to be made.
- Given the critical importance, during any accident or emergency situation, of timely, accurate, and understandable communication to the public, the process for crisis communications needs to be reconsidered, e.g., additional effort might be warranted to prepare emergency spokespersons for conditions associated with the need for current, accurate, and understandable information in a radiological emergency. This is discussed further in Chapter 8.

These types of changes can be accomplished with more frequent internal drills and training and would not be expected to require expanded or more frequent external exercises.

7.3.3 Criterion for Long-Term Habitability (Reentry)

One of the NRC's 16 planning elements for EP is recovery and reentry, which describes decisions and procedures for relaxing protective measures to allow reoccupation of an evacuated area. Though it varies slightly from state to state, most states and the U.S. Environmental Protection Agency (EPA) have a limit of 500 millirem (5 millisieverts) per year as the long-term habitability (i.e., return home) criterion, meaning evacuees from inside the 10-mile EPZ and those outside the 10-mile EPZ who might be relocated are generally allowed back into their homes, businesses, etc., if their radiation dose is projected to be less than 500 millirem (5 millisieverts) per year. The International Committee on Radiation Protection (ICRP) recommends a range of 100 to 2000 millirem per year (1 to 20 millisieverts per year) as a relocation criteria. The Japanese government has set guidelines that allow residents to return to evacuated areas if the projected dose is below 2000 millirem (20 millisieverts) year, the maximum level recommended by ICRP, although this limit is under evaluation at this time.

The reentry limit is key to the potential long-term economic impact of land contamination. The latent cancer fatality risk models, which are central to setting this long-term habitability limit and to other related low dose targets and limits, are based in large part on data from studies that extrapolate the observable effects of high doses and assume the same linear relationship applies to low doses for which there are no observable effects. The main such study is a decades-long, still-ongoing evaluation of the cancer incidence in Hiroshima and Nagasaki atomic bomb survivors.

As a result, the health risk from low-level radiation is uncertain and difficult to characterize. Opinions on how to treat this health risk vary, ranging from the linear, non-threshold (LNT) approach, in which it is assumed that risk is linearly proportional to dose at all levels of radiation and there is no dose below which cancer risks are non-existent, to assuming no cancer risk for doses below a level equal (in the U.S.) to an average natural background combined with average annual medical exposure [~ 620 millirem (6.2 millisieverts) per year] or in the case of the Health Physics Society (HPS), no cancer risk for doses less than 5 rem (50 millisieverts) per year with a lifetime limit of 10 rem (100 millisieverts) [77]. In fact, there is a growing body of knowledge asserting that the assumed linear relationship between dose and effect is not valid.[5]

[5] Recently, researchers with the U.S. Department of Energy (DOE)'s Lawrence Berkeley National Laboratory have found evidence that cancer risks might not be directly proportional to dose. Data show that at lower doses of ionizing radiation,

The long-term habitability (return home) limit is becoming a significant issue for the Japanese government, with regard to the timeline for return of residents evacuated from the 20-kilometer zone around the Fukushima Dai-ichi plant. For the Japanese, the question of how to respond to uncertainties in risk from low-level radiation is as much an ethical one as a scientific one, i.e., the tradeoff of this scientific risk with the socio-economic costs of extensive cleanup or the cost of not allowing people to return home to resume their normal lives.

There is a significant need for better science on long-term health consequences of low-level exposure to radiation. Accordingly, the ASME Task Force strongly urges that ongoing work on low-level radiation effects be expanded by the global nuclear industry and the health physics and radiation biological-effects communities, to include gathering and applying data from the Fukushima accident, to improve the science of low-level radiation risk. Also, an international decision framework and standards for return-home and long-term habitability should be developed.

7.3.4 Building Public Trust

In the aftermath of the Fukushima Dai-ichi accident, the global nuclear power industry has a unique opportunity and a critical obligation to improve public trust in nuclear power as a safe and secure means of generating electricity. EP is often viewed as the principal public face of a nuclear plant in the surrounding community. Further discussion of building public trust, both broadly and in the context of EP, is provided in Chapter 8.

7.3.5 Updated Basis for EPZ Size

This section briefly reviews the technical basis for the current EPZ size in the U.S. and the margin and conservatism that are inherent in this EPZ size. Also discussed is the need for a globally-coordinated effort to update the technical basis for EPZ size.

7.3.5.1 Basis for Current EPZ Size

The technical basis for the EPZ distance associated with the plume exposure pathway, currently 10 miles (16 km) in the U.S., and the EPZ distance for the ingestion exposure pathway, currently 50 miles (90 km) in the U.S., is contained in NUREG-0396 [78], which was published in 1978 by a joint NRC-EPA Task Force.

In determining the recommended 10-mile plume exposure EPZ, four considerations were addressed in NUREG-0396. These considerations were later restated in NUREG-0654 [79], the joint FEMA-NRC document that prescribes implementing guidance for EP actions onsite and offsite.

a. Projected doses from design basis accidents (DBAs) would not exceed the protective action guide (PAG) levels, i.e., 1 to 5 rem total effective dose equivalent (TEDE), outside the zone.

b. Projected doses from less-severe (i.e., most) core-melt accidents would not exceed the PAG levels outside the zone.

c. For more severe (worst) core-melt accidents, immediate-life-threatening doses would generally not occur outside the zone.

d. Detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event this proved necessary.

DNA repair mechanisms work much better than at higher doses. This non-linear DNA damage response casts doubt on the general assumption that any amount of ionizing radiation is harmful and additive. The significance of the work being conducted at Lawrence Berkeley and other facilities is that there are encouraging signs that the foundation is being laid to establish exposure limits based on a sound understanding of the hazard radiation represents.

In evaluating the projected dose levels and associated probabilities, the frequencies and consequences of severe accidents used in NUREG-0396 came primarily from the Reactor Safety Study (WASH-1400) [32]. This report, published nearly 40 years ago, contained the first nuclear plant PRAs performed in the U.S. and reflected the perspectives and state of knowledge on severe accidents that existed in the early 1970s. WASH-1400 contained accident source terms (i.e., radioactivity-release magnitude and timing from accidents) that are now known to be unrealistic (i.e., releases that are much larger and faster than what is predicted today).

The ingestion-exposure EPZ is based on potential ingestion doses to the thyroid through the cow/milk pathway. NUREG-0396 recommended 50 miles based on the observation that this planning basis for milk ingestion would approximately correspond to the 10-mile plume-exposure distance. That is, given a core-melt accident, there is about a 30% chance of exceeding the 1-rem TEDE PAG at the 10-mile plume-exposure EPZ, and a 30% chance of exceeding the 1.5-rem milk-pathway PAG at the 50-mile ingestion-exposure EPZ.

7.3.5.2 Conservatism in EPZ Size

A reassessment of EPZ size, using more recent and realistic severe accident source-term information, indicates there is significant margin and conservatism in the 10-mile plume-exposure-pathway EPZ for typical U.S. operating plants. [80] This is shown in Table 5, which contains conditional probabilities of dose exceedance for 1 rem, 5 rem, 50 rem, and 200 rem. The second column is for recent source term information at a distance of 2 to 3 miles (3.2 to 4.8 km), and the third column is from NUREG-0396, Figure I-11, which is based on WASH-1400 source terms and is for 10 miles (16 km). Table 5 clearly shows the margin and conservatism that exist in the 10-mile plume-exposure-pathway EPZ. Margin is also expected for the 50-mile ingestion-exposure-pathway EPZ. Based on this information, for typical U.S. operating plants, existing EPZ size is expected to fully satisfy public-health-and-safety-protection guidelines. There would be even more conservatism for Generation III+ plants now starting construction in the U.S. and around the globe.

Table 5 – Dose Exceedance Results for Plant with Recent Source Terms vs. NUREG-0396

Dose	Conditional Probability of Dose Exceedance	
	Typical U.S. Operating Plant with Recent Severe Accident Source Term Information (2-3 miles, 3.2-4.8 km)	NUREG-0396, Fig. I-11 (10 miles, 16 km)
10 millisieverts (1 rem)	0.1	0.3
50 millisieverts (5 rem)	0.07	0.25
500 millisieverts (50 rem)	0.02	0.1
2 sieverts (200 rem)	<0.001	0.001-0.01

7.3.5.3 Need for Updated Technical Basis for EPZ Size

While NUREG-0396 reflected the state of knowledge on severe accidents and the prevailing view on risk that existed in the 1970s, for reasons discussed below, it is appropriate to consider an update to the basis for EPZ size. This is not necessarily directed at reducing EPZ size; rather it is directed at application of state-of-the art severe accident and source term methods to justification of EPZ size.

The primary reason for considering an update is that in the over three decades since the existing EP basis was developed in NUREG-0396, the global nuclear power community has acquired a greatly-expanded operational experience base and a significant experimental and analytical knowledge base on severe accidents. It is now recognized that the relatively-rapid, massive, fission-product releases and severe-accident risks estimated in earlier studies, such as WASH-1400, which was the basis for NUREG-0396, are unduly pessimistic. Use of state-of-the-art scientific information in the basis for EPZ size is consistent with the proposed new safety construct.

A second key reason for considering an update to the basis for EPZ size is the ability of plant operators to prevent and mitigate severe accidents through accident management, which typically has not been credited in PRAs. As demonstrated in the SOARCA study [77], effective accident management provides much-improved capability to manage a range of accident scenarios and substantially decreases the risks of core damage and radioactivity release in a severe accident.

Finally, an updated approach to the basis for EPZ size should strengthen public confidence in nuclear plant safety and EP. The existing basis for EPZ size in NUREG-0396 overstates the risk from nuclear plant accidents and could result in unwarranted actions that poorly serve radiological protection of people. Properly formulated, an updated approach could incorporate the notion advanced by the NRC over the last decade that accident and risk information should be developed and documented in a manner that discourages misuse of the results. Calculation or promulgation of disastrous public health effects or massive releases for highly improbable or unrealistic events helps no one, wastes resources, and frequently results in unfounded fear.

The occurrence of the accident at the Fukushima Dai-ichi plant does not change the need for an updated approach to the basis for EPZ size. EP is intended to address public health and safety. While the Fukushima Dai-ichi accident resulted in significant damage to the plant and significant offsite economic effects in the form of large areas of contamination and dislocation of people from their homes, schools, and workplaces, the effect on public health and safety from radiation was minimal, and the fission product releases were relatively slow and amenable to protective measures.

While the foregoing discussion on the need to update the technical basis for EPZ size is U.S.-centric, consistent with the guiding principles of the New Nuclear Safety Construct, i.e., provide a platform for continued and expanded safe and reliable operation of nuclear power plants worldwide, such an update of the basis for EPZ size should be a globally-coordinated effort.

The basis for establishing the EPZ size for new nuclear plants also needs to be established with appropriate consideration for increased safety of these plants. The overall core-damage frequency calculated for new plants is as much as two orders of magnitude lower than the core-damage frequency for currently-operating plants. Several of the designs for new plants include passive safety features that do not rely on availability of AC power and negate the need for any operator intervention to manage accidents for at least 72 hours. The long lead time available for responding to potential accidents, and the increased safety associated with the passive nature of these designs, suggests there is a technical basis for reviewing the EPZ size for these plants.

In addition, work is ongoing on the part of NRC and industry to address EP for small modular reactor (SMR) designs. NRC has issued SECY-11-0152 [81] which discusses the staff's intent to develop a technology-neutral, dose-based, consequence-oriented EP framework for SMR sites that takes into account the various designs, modularity, and colocation, as well as the size of the EPZ. The industry also is developing a framework for a graded approach to EP and EPZ size binning for SMRs. [82][83]

7.4 Risk-Informed, Performance-Based Approach to EP

EP would benefit from a more risk-informed, performance-based approach for defining requirements and performing regulatory oversight, particularly for new nuclear plants, but also for operating plants. This would be in contrast to the current deterministic approach in the U.S., which was largely developed over three decades ago after the TMI-2 accident and in which there is no clear way to determine the value of a planning element or changes and additions to planning elements. In addition, under the current deterministic approach a licensee has little flexibility as to how to implement EP requirements, despite potentially-significant differences from one site to another, one plant design to another, or both. NRC is undertaking an initial effort to determine feasibility and direction of risk-informing regulatory oversight of EP. [84] The SOARCA Project could also provide the basis for consideration of changes to EP. Development of a risk-informed, performance-based approach that makes EP more cost-effective and increases flexibility for licensees is consistent with a recent series of Presidential Executive Orders and related guidance documents, which state in general that U.S. federal regulations must be based on the best available science; promote predictability and reduce uncertainty; identify and use the best, most innovative, and least burdensome tools for achieving regulatory ends; and take into quantitative and qualitative account benefits and costs. [85]

In a risk-informed, performance-based approach, the requirements could be a set of criteria or targets to be met by the licensee, with associated metrics monitored to confirm that the criteria or targets are met. For example, it could be required that an EPZ and associated emergency plan exist, with predefined emergency action levels (EALs) and associated protective actions. Other possible elements of the EP program would be determined on a site-specific basis by the risk-informed, performance-based system. The risk-informed, performance-based criteria or targets might be that the plant be designed and operated such that accidents are mitigated and there is adequate protection of the public for an appropriate set of severe accidents. Defense-in-depth, which is application of deterministic principles to account for uncertainties, would remain a fundamental part of the requirements developed under the risk-informed, performance-based system, by virtue of the requirements that an EPZ, EALs, and protective actions exist, and by applying appropriate margin in the source terms for the set of accidents being considered. In addition, and consistent with the proposed new safety construct, there is a need to determine, beyond the adequate-protection criteria, whether additional improvements should be implemented based on avoidance of large socio-political and economic impacts.

A risk-informed, performance-based approach to EP in the U.S. would be a multi-year effort that would include pertinent new NRC policy and extensive industry/stakeholder involvement and could lead to a consistent set of risk-informed regulations. However, the state-of-the-art of nuclear plant severe-accident analysis, which now includes integrated treatment of core-damage progression, fission-product transport, accident management, offsite consequences, and emergency response, has matured to the point that the nuclear community has the technical capability to risk-inform EP. This will provide even greater assurance that adequate protective measures can and would be taken in the event of a radiological emergency, while providing better use of EP resources.

7.5 Summary Comments

In summary, the ASME Task Force has the following observations on Emergency Preparedness. Pertinent conclusions are summarized in Chapter 9.

1. The EP infrastructure onsite and offsite should be modernized to be more robust and flexible to provide high assurance that EP-related systems will be functional under unexpected conditions that could cause the accident in the first place (e.g., conditions brought about by events such as extended SBO, including loss of all AC and DC power, damaged site perimeter, and temporary loss of site access).

2. EP exercises and training should be based on more-realistic, slower-developing accident scenarios and accident conditions, such as those resulting from multi-unit events and prolonged SBO events.
3. A better scientific basis should be developed and applied to return-home criteria, including low-level radiation effects, as part of efforts to address effects of land contamination.
4. The nuclear industry and regulators have an opportunity and obligation to improve public trust in nuclear power in the aftermath of the Fukushima accident. With regard to EP, the global public reaction to the accident indicates there is a significant need for improving the science and public understanding of low-level radiation risks.
5. The current EPZ size in the U.S. has significant conservatism and margin. The ASME Task Force notes that for typical U.S. operating plants, current EPZ size should fully satisfy public health and safety protection guidelines. However, the technical basis for the current EPZ size in the U.S. is based on an outdated understanding and description of severe accidents and should be updated to reflect what has been learned over the last 40 years of research and plant operation (lower, slower fission-product release during accidents) and to reflect the increased safety of new nuclear plant designs. Such an update should be a globally-coordinated effort.
6. EP would benefit from a more risk-informed, performance-based approach for defining requirements and performing regulatory oversight.

8 REINFORCING THE NUCLEAR SAFETY CONSTRUCT

8.1 Introduction

This section focuses on the important role played by communication in the way nuclear technology is perceived by the public at large and its influence on the socio-political consequences of accidents.

Community outreach programs are an essential element in deployment of nuclear technology. This is a necessary step in gaining the trust of the public and securing the support of other stakeholders. To reach stakeholders in an effective manner, outreach programs need to be substantial and sustained. Each generation of citizens should be engaged as they move through the educational system, and this engagement must continue through adulthood.

There are two distinct communication elements to community outreach: everyday dialog about the technology to develop informed stakeholders, and crisis communications during an emergency. The daily-information portion of outreach is discussed in Section 8.2 while the crisis-communication portion is discussed in Section 8.3. These communications are far more effective if a comprehensive public-outreach program has been implemented over an extended period. **Knowledge and understanding of nuclear technology is in the best interest of society.** Section 8.3 also describes the onsite and offsite communications infrastructure that will be called upon in the event of an emergency. Communication links to offsite support and federal, state, and local officials are discussed, as well as the importance of an educational program for officials and the public at large, to provide perspective in the event of an emergency.

8.2 Community Outreach Conducted During the Normal Course of Events

Being able to communicate technical information in a transparent and understandable manner is a skill, the importance of which is often underestimated. This skill is not and cannot be limited to media specialists or designated spokespersons, but is particularly important for engineers and technology experts, as they seek to inform the public on the specifics of a complex technology. This is a particular challenge for the nuclear industry, for which the issues are often technical, but also political. Sorting the differences among these diverse aspects, and clearly articulating the facts to stakeholders should be done on a continuing basis, using simple, easy-to-understand language.

Fear of the unknown is a well-known human reaction. Fear of radiation falls into the category of an unseen threat, even when used for medical purposes. This fear arises from the technology's genesis and early development within the nuclear weapons program and its association with adverse long term health effects, i.e., latent cancers. Development and implementation of industry programs communicating the essence of nuclear science and technology help inform the public and other stakeholders—allowing them to make more informed judgments on deployment of the technology. Public understanding is an essential element in bringing the benefits of any technology to society, and this is especially true for nuclear-related technologies.

Sound economic, energy, and environmental policies are important for every country. The science associated with each of these areas is complex. Divergent opinions arise, and the media and public are left in a difficult position in terms of judging who to believe and what course to take. As a consequence, a key element in ongoing communications is to equip the general public and key stakeholders with something more than a rudimentary understanding of nuclear technology.

Informing the public is an obligation of those professionals who understand nuclear technology. As a public service, professional societies, such as ASME, ANS, HPS, and others should encourage and prepare their members to reach out to the general public and media. In addition, the expectations of K-12 education should be elevated with respect to science and engineering. Qualified teachers presenting curricula supported by factually-correct text books should lay a sound foundation in the sciences, preparing students to function in a society increasingly dependent on complex technology. In addition, students should be encouraged to consider careers in science, engineering, and mathematics.

The ASME Task Force has concluded that a better understanding of nuclear power gained by the public in advance, or even in the absence of an accident is also important for public support of normal operations because people with little or no knowledge of nuclear power often have the perception that the risk from nuclear power is higher than it actually is. This unwarranted perception often extends beyond concern for oneself to concern for entire families, particularly children.

Preparing spokespersons for communication with the public, especially during crisis conditions, is vital. Communication of complex issues—especially during an accident—can significantly influence the socio-political impact of the accident in terms of maintaining calm and avoiding psychological trauma. A key element in picking the right spokesperson is the trust that person engenders with the public and the media. In this regard, spokespersons must be knowledgeable, articulate, and trustworthy.

8.3 Crisis Communications: What is Happening and Why

In a rapid or emerging crisis situation, modern media now provide essentially instant communication channels. The connectivity and speed of modern news communications have been revolutionized over the past several decades. The form, content, and pedigree have changed dramatically compared to when there were only a few responsible media outlets for fast-breaking stories. There are multiple television and radio channels that run continuous news, frequently not adequately vetted, because there is not time to do so. Their objective is to get the information out to the public from any source they can find; they are incentivized to "scoop" the competitors with the latest breaking stories. Moreover, chat rooms, bulletin boards, and personal messaging routes can dwarf traditional means and methods of structured interviews, press conferences, and formal media releases. Information is now "free" and cannot be controlled.

During the TMI-2 accident in 1979, communications were difficult, typically with hard wired telephones or personal interviews. The information cycle was tied to the schedules of the major networks, typically on an 8- to 24-hour cycle, depending on when the news was being made. During the Fukushima Dai-ichi accident, information was streamed continuously by whoever obtained access to it, sometimes from amateurs watching television. This information was then streamed over the internet worldwide through various social networks. The ASME Task Force observed that much of the material was not researched or vetted and was subject to the interpretation of the observer, who often was not versed in communications or technically qualified to comment on what was happening. The continuous television news networks picked up video, interviews, or "expert" commentary, and released it to millions of viewers 24 hours a day.

Effective communication both within the industry and with the public was a significant deficiency during the unfolding events at Fukushima Dai-ichi. Inadequate communication led to media inaccuracy and overreaction, as well as a significant erosion of public trust in the industry as a whole. Even when communication of information on radiation exposure and level was attempted, the data were often described in units that were incomprehensible and overwhelming to most members of the media and the public. These communication deficiencies must be addressed as the nuclear industry incorporates the lessons learned from the event and makes improvements.

This new "open" paradigm in media communications requires a concerted, effective, sustained, and strategic program to develop a generation of informed citizens that can cull through the numerous and divergent sources of information and draw valid conclusions. This is not something that can be accomplished overnight. It requires a four-pronged multi-generational effort, beginning with instruction in school grades (K-12), in concert with outreach programs for the general public, relationship building with the media, and interactions with policymakers.

To properly communicate during a crisis, both formal and informal channels must have been established ahead of time, procedures and knowledge-streaming means already developed, with roles and responsibilities agreed. All must be practiced on a regular basis with the responsible people taking their roles of improvising and responding instantly in crisis exercises. This is typical of modern emergency management practices and drills, but these are usually focused on the key crisis management aspects of fires, floods, and earthquakes. There are established requirements by regulatory authorities on how frequently such exercises must be held, but they are not always realistic nor are they always broad enough to test the capabilities of the participants and the procedures used. The time to learn instant messaging properly is before crisis communications are needed; they cannot be executed in real time without proper training. If there is any expectation of effectiveness, subject-matter experts and media commentators must adapt, decide, and debate in a crisis without a script. Thinking on your feet, not giving wrong answers, talking to the right experts, gathering the correct data, making the insightful analysis, and communicating clearly all require team efforts that are more comprehensive and externally connected—basically a virtual combination of "war room," "media center," and "emergency response center."

In a broad-area disaster, such as occurred at Fukushima Dai-ichi, external and internal communications are complicated, because the needed normal physical infrastructure (supplies, roads, railways, power lines, phones, bridges, instrumentation, pumping, etc.) and the social infrastructure (staffing, expert advice, off-site support, authorizations, messaging, management processes and guidance, etc.) can be disabled or disrupted. The equipment and personnel must be able to operate under these severe conditions. In essence, the equipment and the staff must be self-powered at both ends of the communication links and must have sufficient life (e.g., battery capacity and stamina) to operate for the duration of the disaster. Of course, it is possible to "fly in" additional equipment or batteries, as long as access to the site can be acquired during the emergency conditions; nevertheless, there must be sufficient onsite equipment life and staffing to ensure and sustain communications until this support and backup arrive.

8.3.1 EP Command and Control, and Decision-Making Authority

The measures described throughout this document should position a reactor operator or other plant staff to be better able to preclude, interdict, and mitigate the consequences of accidents, by improved design, training, and access to ancillary resources. These enhanced capabilities have value only if appropriate actions are taken to use them in an effective, timely, and coordinated manner. This is the role of Emergency Preparedness (EP) which establishes command-and-control protocols and clearly defines lines of authority.

During an emergency, plant personnel are called upon to act under conditions that are highly stressful and require decisive action. Selection of personnel who have the technical and leadership skills to carry out their duties during routine and off-normal conditions is important. In addition to possessing the proper demeanor and judgment, these individuals need to be trained to respond to unexpected events. Procedures, lines of communication, and organizational structure and authority during accidents need to be developed and understood by operators and supporting staff, as well as by public sector authorities and political leaders.

"Emergency Preparedness" defines the activities that lay the foundation to satisfy these needs and respond effectively to reactor accidents. By implementing the following recommendations, challenging events can be managed in an effective manner.

Lessons from the Fukushima Dai-ichi accident reinforce the need for improved command-and-control strategies and structures in a number of areas, including the following:

- The need for information from the plant, including installed plant instrument readings, and if such instruments are not available or reliable, alternative means such as portable instruments and computational aids;
- The need for a clear approach to transition from EOPs to SAMGs and EDMGs;
- Decision-making during multi-unit events, including understanding effects such as control room configuration (common control rooms vs. separate control rooms) and distance of separation of the units (e.g., in the U.S., large separation at Palo Verde vs. small separation at Indian Point);
- Command and control for prioritizing limited resources;
- The location of decision making (control room, plant, Technical Support Center (TSC), Emergency Operations Facility (EOF), offsite); and
- The need for an approach to command and control, responsibilities, and lines of authority during emergencies that is preplanned for large-scale events and beyond-design-basis events, but at the same time is designed with improvisation in mind to allow for unforeseen events and conditions.

Some of these areas are already being addressed; for example, in the U.S., the NRC's advance notice of proposed rulemaking (ANPR) on "Onsite Emergency Response Capabilities" [86] and the ongoing U.S.-industry response to the ANPR. International perspectives have been documented in IAEA Safety Guide NS-G-2.15 [87]. The ASME Task Force supports these efforts and encourages international cooperation and development of global standards of excellence on command-and-control strategies.

8.4 Earning Public Trust

To the ASME Task Force, the term "public trust" denotes the level and type of vulnerability the public is willing to assume with regard to nuclear power. Today, at least a portion of the public believes that the majority of its vulnerability is involuntary and results from a sizable influence imbalance that enables powerful interests to achieve greater rewards and assume far less risk than the public.

Public trust and confidence are essential if nuclear power is to achieve and sustain social and political acceptance. Prior to the Fukushima Dai-ichi accident, as a result of a sustained period of reliable and safe nuclear power plant operations lasting more than 25 years, public support for continued operation of nuclear power plants in the U.S. reached a new high. Toward the end of 2010 this support reached 62% in the long-running Gallup poll. A "nuclear renaissance" was heralded worldwide, and new plants were in the advanced planning stages.

In the days following the Fukushima Dai-ichi accident, the measured reaction of the U.S. Administration and the assurances by the NRC that U.S. plants were safe helped to maintain generally-favorable public views about the safety of nuclear power, with 47% of the public being positive about nuclear power and 44% being negative. This result compares to polling results before the accident with 53% favorable and 37% unfavorable. [88] However, public distrust and political considerations in other countries—notably Germany and Italy—precipitated government decisions to phase out existing nuclear power plants and/or cancel plans for new plants as a source of electricity. Adopting financial terminology, stress tests were mandated in Europe for all existing plants. In Japan, at the direction of the government, several nuclear power plants were shut down immediately following the accident at Fukushima Dai-ichi, and as of May 5, 2012 none of Japan's 54 reactors was operating, These plants are expected to remain shut down for an extended period, pending safety (stress test) reviews, plant and site upgrades to address revised tsunami, earthquake, and power loss protection criteria, and potentially-protracted approvals by local prefectures. Depending on tradeoffs of the state of public trust for nuclear power in Japan vs. economic hardships associated with the loss of generation, a worst-case scenario would have many or possibly even all of these reactors shut down permanently.

In the U.S., the public trust in nuclear power needs to be strengthened. If trust is further undermined by another major accident with significant environmental consequences, support for continued operation of existing plants and construction of new plants could be at risk.

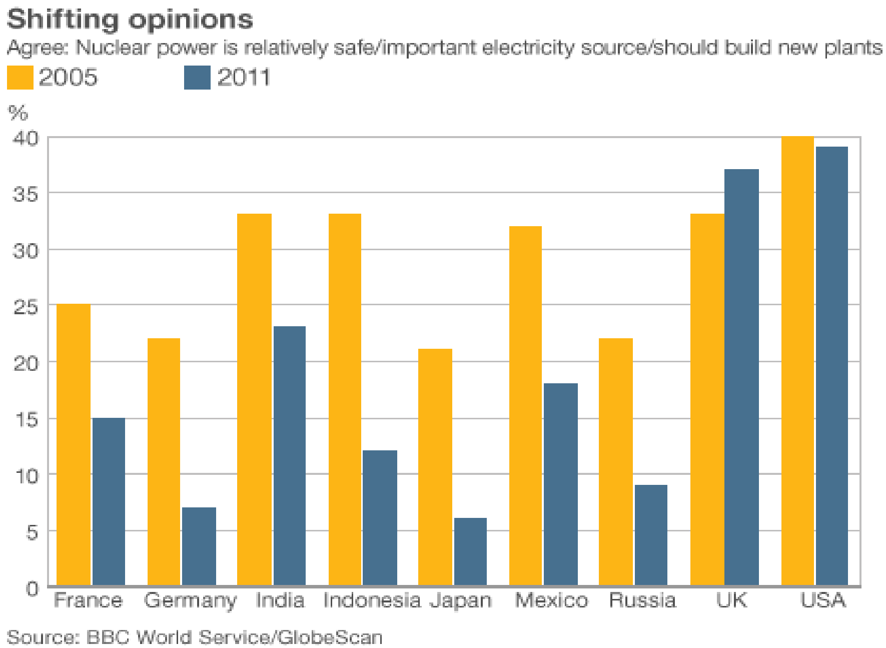

Figure 8 – Public Opinion on Nuclear Power Before and After the Fukushima Accident

Figure 8 shows the polling results from BBC World Service Globe/Scan for 2005 and 2011, i.e., well before the Fukushima Dai-ichi accident and just afterward. The steady polling numbers for the U.S. and the U.K. are in stark contrast to other countries—which show a substantial drop in positive opinion of nuclear power. This poll broadened the questions asked to include not only the relative safety of nuclear power plants, but also the importance of nuclear power as an electricity source and whether or not new nuclear power plants should be built.

In principle, trust creation is a process in mutual value creation among parties that are unequal with respect to power, resources, and knowledge. [89] This process involves the following:

- Creation of mutually-shared values among stakeholders (examples include fulfilling the need for clean energy and socially responsible behavior by the organizations associated with the business of nuclear power);
- A balance of power, wherein the risks and opportunities are shared to a degree (examples might be the perceived fairness of the price of electricity and the openness, integrity, and transparency with which the organization operates); and
- Existence of regulatory and other safeguards that limit vulnerabilities and prevent one party from imposing its will on, or overpowering the interests of, another party.

Public trust cannot be assured exclusively through better or safer nuclear power technology and better public education, as has been the approach in the past. Rather, it must be recognized that public trust emanates from trust in the organizations and the people who design, build, operate, and regulate nuclear power. Professor Michael Golay of MIT correctly notes [90] that social acceptance of nuclear power rests on a foundation of

- The technology itself must perform reliably and safely,
- A shared belief that the technology is mutually beneficial, and
- Trust in the integrity, competence, and performance of those designing, building, operating, and regulating the technology.

With respect to the first point, the U.S. nuclear industry, led by utility operators, has dramatically improved its operating record over the decades since the TMI-2 accident in 1979 (see Chapter 2, Figure 4 for Capacity Factors and Significant Events over the past two decades).

Certainly, there continues to be a role for safer nuclear power technologies (e.g., passively-safe plants with improved man-machine interfaces), but such steps are primarily to reduce the incidence and severity of accidents. New technology is an important, but not sufficient, component of increased public trust.

People living near an operating nuclear plant are more likely to share a belief that the technology is mutually beneficial and that there is a basis to trust those associated with operation of the facility. Investments in educational and other valuable activities enhance the lifestyle within the community where these plants are sited. In addition, operators and other plant employees are friends and neighbors, not faceless corporate representatives who live far away.

The importance of trust in the organizations and people who design, build, operate and regulate nuclear power is clear. Steps must be taken to build and sustain a more robust public trust—one that has a foundation that can weather the ups and downs that every technology encounters. New avenues for trust-building activities need to be established; old ones need to be enhanced. This requires a broad-based approach from all parts of the nuclear industry; from university professors who teach the new generation of engineers and industry professionals to the companies and organizations that constitute the working nuclear industry.

Building a solid foundation for sustained public trust requires increased personal and organizational commitments to demonstrating competency, transparency, integrity, and social responsibility, with all stakeholders reaching out in a personal way—both within and beyond the communities where plants are located.

Professional societies like ASME already enjoy a high degree of public trust based on their professional and scholarly services and the perception of being relatively free from conflict of interest. This provides the opportunity for ASME and other professional societies like ANS and HPS to be more active in their public education activities, as a source of unbiased information, and to encourage their individual members to increase their trust-building activities as a key element of their professional and ethical responsibilities. Increasing the Societies' neutral convening activities by creating opportunities to listen to the broader public, including people and groups both opposed and in favor of nuclear power, will help improve communication and understanding. Finally, ASME's regular contact with socio-political leaders and the media will position the Society as a reliable technical resource when nuclear power issues are in the news.

8.5 Summary Comments

The ASME Task Force emphasizes that the socio-political impact of an accident at a nuclear reactor is a function of more than the physical events occurring at the plant. The socio-political consequences are no less determined by the *perception* of the threat the accident poses for the health and welfare of the individuals living and working in the vicinity of the facility and the manner in which crisis communications are carried out.

A concerted, sustained, outreach effort is essential to ensure that the public is informed on the essentials of the technology and has an accurate understanding of potential hazards and associated risk. This educational outreach effort addresses public fears with respect to the essence of the technology and prepares them to keep events in perspective.

Finally, sound communications will be better received if a level of trust exists between the communicating parties. Trust must be earned. It is a painstaking process and can be quickly lost. In the end, personal integrity and competent staff striving for excellence in operations should ultimately engender trust and confidence among the public.

9 FORGING A NEW NUCLEAR SAFETY CONSTRUCT

9.1 Introduction

The ASME Presidential Task Force on Response to Japan Nuclear Power Events commenced its review and assessment of the Fukushima Dai-ichi accident and their global impact in September 2011. At the time, significant information on the accident was already available, and the flow of information continues after the one-year anniversary of the accident. The relevant information released from multiple national and international cognizant organizations, and referenced herein, served as the basis for the ASME Task Force review and findings. Although not yet complete, the already-reported information appears consistent and identifies causes and lessons learned from the accident.

The Fukushima Dai-ichi units are the first nuclear reactors in the world to experience core degradation due to a catastrophic external event, and the first light-water-reactor plant to experience a multi-unit accident resulting in large radioactivity release to the environment. The severe accident at the Fukushima Dai-ichi plant has and could further affect energy portfolios worldwide, by skewing the overall perception of nuclear electricity generation, with disregard to its overall safe record of reliable day-to-day operations, as well as its beneficial impacts on fuel diversification, energy security, climate change initiatives, and stability of electricity costs. For this reason, the ASME Task Force has concentrated on understanding core damage accident progressions, their impact on society, and potential improvements in preparedness, accident management, and protective measures. The ASME Task Force also sought to use the body of knowledge from previous accidents and knowledge emerging from the Fukushima Dai-ichi accident to generate a comprehensive review and associated recommendations that would address broader global concerns. The overall intention was to arrive at conclusions that would be beneficial to society and of significant use to the global nuclear industry response to the accident, focusing on nuclear safety improvements that specifically address the Fukushima Dai-ichi issues.

The major concern during the Fukushima accident and for many months afterward was the radiological health and safety of the public, a grave concern that deserved the full attention of Japan and the support of the global community. The Fukushima Dai-ichi nuclear accident, from a radiological perspective, resulted in no prompt fatalities, and there is a continuing expectation of no significant delayed radiological public-health effects. [3] [6] [15] This favorable public health outcome is due to several factors, including the emergency protective actions, the prevailing winds blowing radioactive plumes to the open ocean, and the inherently-slow radioactive releases from nuclear accidents. For the millions of people around the world who followed the media coverage of the accident, the sustained reality of the absence of radiological health effects should become an important issue for the discussion of the hazards and consequences from nuclear power accidents.

While public health and safety will continue to be the dominant safety criteria for nuclear power plants, there are other, significant consequences of nuclear power plant accidents that deserve serious attention and are the intended focus of this report. In the case of the Fukushima Dai-ichi accident, these other significant consequences include: radiological contamination of a large populated area in Japan, initial relocation of more than 100,000 people for radiological protection, broad psychological stress on the Japanese people, the loss of economic productivity of the contaminated areas until remediated or deemed safe, wholesale curtailing of nuclear power generation across Japan, and accompanying substantial economic impact in Japan and other countries. It is this contrast between the lack of radiological public health effects and the substantial societal impact from the Fukushima Dai-ichi accident that claimed the attention of the ASME Task Force and resulted in the main thrust of its proposed approach for forging a new nuclear safety construct.

9.2 The Evolving Nuclear Safety Construct

From its origins, civilian use of nuclear materials has been based on establishing an effective safety construct for protection of the public and workers against radiation hazards. This is as true for nuclear medicine as it is for nuclear power. A construct incorporates all of the industrial and regulatory frameworks necessary for effective execution of the functions essential to the desired outcomes. Organization and coordination are crucial to the functioning of a construct. A safety construct is defined by the expected outcomes. The complexity of the nuclear power infrastructure has required an equally-complex safety construct that includes an independent regulatory framework.

While the details may differ from country to country, the existing safety construct for nuclear power generally has the following features throughout the world: structures, systems, and components specified by the design basis; additional equipment capable of supporting safety functions; an extensive regulatory framework addressing protection of public health and safety, the environment, and common defense and security; all pertinent federal, state, and local laws and rules, and the supporting organizations; and technical and accident management structures necessary for design, construction, operation, and maintenance of the plant. As discussed in this report, the existing nuclear safety construct has evolved, as needed for safe and reliable operations and as required by rules and regulations.

While many changes have occurred due to cumulative operational-safety experience, a series of significant changes were the direct consequence of lessons learned from accidents. The lessons learned from the TMI-2 accident are a prime example of the substantial transformation of the nuclear industry after a severe accident, with major changes driven by both regulatory requirements and the industry drive for achieving excellence in operational safety performance through the voluntary creation of INPO. TMI-2 lessons learned prompted substantial equipment, accident management, and emergency preparedness improvements in the U.S. and globally. The Chernobyl accident had a similar impact in Europe; its impact in the U.S. was less because of the on-going changes from the TMI-2 accident, and because of the significant differences between the U.S. and European infrastructures. The accident at Fukushima Dai-ichi is different in many ways from these earlier accidents, with broad global applicability, and will surely result in many substantial changes to the existing safety construct.

Indeed, the response of regulatory authorities and the nuclear industry worldwide to the Fukushima Dai-ichi accident has been rapid and thorough, and is continuing in earnest. The reason is that, in many respects, the Fukushima Dai-ichi accident has important differences from the other accidents, yet similarities as well. To prevent, interdict, or mitigate potential vulnerabilities to extreme external events, both the differences and the similarities to previous reactor accidents should be addressed. The key difference is the realization that a rare yet credible external event can lead to an extended loss of offsite power, station blackout, and/or severe damage to onsite electrical distribution systems and ultimate heat sink, leading to loss of reactor cooling and containment integrity. As explained in this report, there are multiple, interdependent issues present at Fukushima that eventually led to the meltdown of multiple cores and release of radioactivity to the environment, including design basis, accident management, and emergency preparedness issues. Also contributing was the confusion caused by symptoms misinterpreted as degradation of a spent fuel pool, which has led to the realization that better instrumentation is needed for spent fuel pool monitoring and performance assessment. However, the fundamental issue remains that Fukushima Dai-ichi was neither designed nor prepared to handle the earthquake-tsunami combination that was unleashed on its site. Therefore, the importance of extreme external events with high consequences has taken a front position in the approach to implementation of lessons learned from the events in Japan.

The combination of worldwide reviews and analyses of the events at Fukushima, and in particular, efforts to address the key set of safety lessons learned in the U.S., have already begun to establish an evolving safety construct that should provide improved defenses against rare yet credible events that could threaten nuclear power plants, regardless of the cause. The ASME Task Force approach in this work is to draw on these important lessons learned, to provide guidance in proposing the new safety construct discussed below.

9.3 The Lesson Learned

As discussed throughout this report and highlighted above, there is already a set of important lessons learned from the Fukushima Dai-ichi accident; this set of lessons learned is the main focus of the global safety response to the accident. Notwithstanding this fact, the ASME Task Force focused on addressing an overarching lesson that is central to the need for systematic actions in response to the consequences of the accident caused by the Great East Japan Earthquake and Tsunami, as highlighted below.

The Overarching Lesson Learned

Protection of public health and safety from radiological releases has been and continues to be the primary focus of nuclear safety. The present body of knowledge, including lessons from severe reactor accidents, establishes the importance of maintaining that focus, yet brings out a relevant fact:

The major consequences of severe accidents at nuclear plants have been socio-political and economic disruptions inflicting enormous cost to society.

As noted above, the multiple-reactor Fukushima accident has had no significant radiological consequences to the population. Thus it could be concluded that the ultimate safety goal of providing radiological protection of the public has been met. However, this would be an overly-narrow conclusion: three reactor cores suffered significant fuel melting, resulting in large uncontrolled radioactive release to the environment. Thus, the governing reactor safety criteria—no fuel melting and no uncontrolled radioactivity releases—were not met, and this is not an acceptable outcome. Moreover, other severe societal consequences ascribed to the Fukushima Dai-ichi accident, as partially described in Section 1.3 and Appendix A.2, deserve thoughtful attention. The socio-political and economic consequences of the accidents at TMI-2, Chernobyl, and now Fukushima Dai-ichi have affected the energy arena and its political landscape. More importantly, the direct impact on people displaced from work and home, and the overall cost to society require that additional steps be taken on a global basis to ensure that prevention, interdiction, and mitigation of nuclear accidents are effected with a more comprehensive and coordinated approach. Preventing social disruption is essential to earning public trust in nuclear power generation.

The ASME Task Force recognizes the global efforts and effectiveness of regulatory authorities, as well as reactor owners and operators, to protect public health and safety and the environment. The established safety record is enduring. Yet this safety record is now subjected to post-Fukushima Dai-ichi scrutiny and it is expected that regulatory authorities will make changes to the regulatory framework to ensure that an extended safety design basis provides additional protection for the public and the environment. Regulatory requirements establishing a new level of adequate protection are under development; these will address the safety requirements appropriate for regulation, and in accordance with the law.

In addition to the changing regulatory requirements, the owners and operators, who have the ultimate responsibility for safety and protection of property and the environment, are developing additional improvements to their plants and procedures. Nevertheless, it is the view of the ASME Task Force that the nuclear industry can and should do better to protect the socio-economic order and do so without additional regulatory requirements. Multiple cognizant industry organizations, primarily those with a strong focus on nuclear power, should forge an objective global standard for preventing, interdicting, and mitigating severe accidents that prevents or minimizes major impacts on society due to large radioactive releases, using an all-risk approach. This is not a new conclusion regarding the responsibility of the nuclear industry. A similar conclusion was reached in the aftermath of the TMI-2 accident, and the industry rose to the occasion with the creation of INPO. To borrow words from the NRC's TMI-2 Lessons Learned Task Force, "if the basic responsibility for public safety is to remain in the private sector, in the hands of individual licensees for commercial power plants, then…" they need to demonstrate their commitment to go beyond regulatory requirements and address a new nuclear safety construct. [91]

9.4 A New Nuclear Safety Construct

The ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events proposes formulation of a new nuclear safety construct to better serve society and to provide a platform for the continued safe operation and growth of nuclear power worldwide.

A New Nuclear Safety Construct

The set of planned, coordinated, and implemented systems ensuring that nuclear plants are designed, constructed, operated, and managed to prevent extensive societal disruption caused by radioactive releases from accidents, using an all-risk approach.

In standard nuclear terminology, the proposed New Nuclear Safety Construct is inclusive of the existing and evolving construct—yet it reaches beyond adequate protection of public health requirements and established industry best practices. The goal is to ensure that on a global basis, owners and operators implement and sustain the capabilities to protect society from socio-political and economic consequences from a severe accident. **The New Nuclear Safety Construct is intended to achieve the purpose of preventing, interdicting, and mitigating accidents and large releases of radioactivity using an all-risk approach. It is to be implemented by assuring purpose, completeness, and coordination of existing and proposed safety functions for design-basis and beyond-design-basis accidents and accident management phases.**

The U.S. nuclear industry has been addressing the need for operational safety "*regardless of cause*," as have many other organizations. Concurrently, the New Nuclear Safety Construct is proposed to be based on an all-risk approach, addressing a broad range of challenges to nuclear power reactor safety. These challenges should be addressed in a risk-informed manner, with the requisite defense-in-depth, for design-basis events and events exceeding the design basis, to include rare yet credible events; challenges to plant safety should be informed by the capability to mitigate their consequences. In particular, cliff edge effects for credible events and scenarios should be explored, and pertinent mitigation approaches should be implemented.

9.5 Principles of the New Nuclear Safety Construct

The ASME Presidential Task Force had the benefit of several extensive analyses of the TMI-2, Chernobyl, and Fukushima Dai-ichi accidents, and opted not to conduct an additional review, but rather to focus on defining key aspects of the new safety construct. The ASME Task Force also realized that implementation of the safety construct is dependent on individual nuclear plant's design and site characteristics, as well as individual countries' regulatory and other key infrastructures. In other words, the work of assembling the details of, and implementing, a functional safety construct should begin when this report ends.

Key aspects of the construct are the following: the structures, systems, and components in the design basis and extended design basis directly responsible for and capable of preventing, interdicting, and mitigating accidents (whether fixed or movable, on-site or off-site); human performance management; accident management; emergency-preparedness management; communications; and public trust issues. These key aspects are covered in the report with different degrees of specificity in accordance with the set of principles summarized below.

- The ASME Task Force has not addressed specific equipment or system options to satisfy the need to meet seismic and flooding challenges, SBO, electrical distribution and ultimate heat sink survivability, and other matters under consideration as part of lessons learned. The ongoing work by the global nuclear community will be better suited to make such determinations.
- The equipment capabilities under consideration in the U.S. by the industry's FLEX approach, associated U.S. NRC requirements, as well as some bunkered systems in Europe, could be sufficient to satisfy the to-be-determined final elements of the new safety construct, provided the overarching lesson learned is addressed.
- The ASME Task Force has specific recommendations in the areas of extending protection against credible severe events by appropriate complements to human performance and decision-making, accident management, and emergency preparedness. These are listed in Section 9.6.
- The ultimate responsibility for safety rests with the owners and operators of nuclear facilities throughout the world. Fukushima Dai-ichi reinforces the fact that there are limitations to the capability of regulators to implement the level of safety for rare yet credible events with high consequences. Therefore, the capabilities of owners and operators and supporting organizations need to be brought to bear on implementing safety measures beyond regulatory requirements, on a global basis, as appropriate.
- The new safety construct relies on the fact that owners and operators can be effective beyond regulatory frameworks. By avoiding the significant consequences that occurred at TMI-2, Chernobyl, and Fukushima Dai-ichi, implementation of the new safety construct would serve society and owners and operators in essential ways.
- To forge a new safety construct for nuclear power plants, one that builds upon the already-substantial existing safety construct and the improvements being made, the approach must be consistent with practical and achievable improvements to reactor safety and radiological protection.
- The approach must be focused on achieving support from multiple, cognizant industry organizations, on a global basis, able to foster the principles outlined in this report and contribute to forging a new safety construct that would better serve society. These organizations should include owners and operators, owners' groups, reactor and equipment vendors, architect/engineering companies, national and international expert organizations, regulators, and consensus nuclear standards organizations.

9.6 Additional Conclusions on the New Safety Construct

Additional conclusions on the safety construct are summarized below.

9.6.1 Design Basis Extension

It has been noted that some parts of Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," relevant to the accidents at TMI-2 and Fukushima Dai-ichi (e.g., common-cause failures and combinations of events) were not completed. Such revisions are currently under consideration by the NRC and the nuclear industry. Others deemed necessary to complete Appendix A also should be undertaken, including risk-informing appropriate criteria. By updating the General

Design Criteria, a better design basis will be established to enhance the safety of both operating and new plants.

9.6.2 Risk-Informed Defense-In-Depth Construct

The construct must be predicated on all modes of plant operations, all-risk, full-scope risk assessments, including PRA Level 3 (consequence) analysis, that are well integrated with deterministic approaches, to achieve a greater level of defense-in-depth for all nuclear power plants. Owners and operators can benefit from such tools in making decisions on which strategies are most effective. To maximize the improvements dependent on full-scope risk assessments, generic, high-level safety goals for new plants should be agreed to internationally. It is duly noted that "A Proposed Risk Management Regulatory Framework" was just released by a task force headed by NRC Commissioner George Apostolakis; it presents valuable findings, and should certainly become an intrinsic component of the risk-informed, defense-in-depth construct. The ASME Task Force agrees that "risk assessments can inform decisions about appropriate defense-in-depth measures." [38]

9.6.3 Human Performance Management

To reduce the propensity, rate, and opportunity for human error, and to significantly improve safety and reduce risk of modern complex system failures, there must be an increased focus on the human-performance aspects of decision-making and its management, before, during, and after occurrence of rare yet credible events.

9.6.4 Accident Management

A uniform global standard of excellence for reliable protection against extreme external events, which would include changes to prevention, interdiction, and mitigation capabilities of a nuclear plant, must include corresponding accident-management capabilities. The global community should define the level of extreme external events against which plants, workers, and the general public must be protected. "Protect" is meant to include reliable accident management measures in place, either as part of the design basis or as part of a pre-determined coping capability to deal with the defined level of external events, with due consideration to regional characteristics.

9.6.5 Emergency Preparedness Management

The EP infrastructure onsite and offsite should be improved to be more robust and flexible, to provide high assurance that EP-related systems will be functional under unexpected conditions, including multi-unit events and prolonged SBO events. In addition, EP exercises and training should be based on more-realistic, slower-developing accident scenarios and accident conditions, such as those resulting from multi-unit events and prolonged SBO events.

The technical basis for the current EPZ size is based on an outdated understanding of severe accidents and should be updated; the update should be a globally-coordinated effort.

The provision of EP would benefit from a more risk-informed, performance-based approach for defining requirements and performing regulatory oversight. The ongoing international controversy indicates that refined definition of reentry conditions to areas affected by radioactive releases should be supported by improved science on low-level radiation effects.

9.6.6 Communications and Public Trust Issues

Enhanced communications are needed during normal and crisis situations, including consideration of timely informing of the diverse array of nuclear-power stakeholders. It is too late to try to determine how to properly communicate after a crisis situation emerges. Therefore, to ensure stakeholders can rely on solid factual information, communication protocols must be established and maintained.

A renewed and expanded effort is required by industry, government, and appropriate professional societies to build and earn public trust in nuclear power, including a strong emphasis on the element of *personal trust* and transparency between the public and the people who design, build, operate, and regulate nuclear power plants.

9.7 Recommendations for Next Steps in Forging a New Nuclear Safety Construct

ASME is a professional engineering society with more than 127,000 members in over 140 countries and is a developer of 500 standards and related conformity assessment programs that are widely used around the globe in nuclear power and many other industries. With this capability and experience, ASME can launch appropriate actions and communicate with other professional societies, industry organizations, and government agencies worldwide in defining and supporting implementation of global actions to prevent and mitigate the consequences of severe nuclear accidents, in a manner similar to ASME efforts a century ago in addressing widespread boiler explosions. Such actions should be taken in three steps: (1) convening workshops of key global nuclear stakeholders to arrive at a consensus for how best to move forward in forging a new nuclear safety construct; (2) standards development; and (3) dissemination of the conclusions and guidance from this work in a comprehensive spectrum of media. However, ASME can only initiate discussion and try to help build consensus; it is up to the nuclear industry and regulators to implement the elements of the New Nuclear Safety Construct.

9.7.1 Workshop(s) on Forging a New Nuclear Safety Construct

ASME should sponsor a workshop or series of workshops to assemble a set of consensus recommendations that can be endorsed by key global stakeholders as soon as practicable. There are common views—yet differences of opinion—among stakeholders, concerning how the global nuclear power industry should proceed following the Fukushima Dai-ichi accident. ASME is well-positioned to act as an unbiased broker to convene workshops or meetings at which stakeholders can reconcile those differences. It is likely that follow-up activities (e.g., joint industry or government-funded projects and focused single-topic workshops) will be needed to develop detailed recommendations. However, there is a need to start forging a new nuclear safety construct by communicating the basic elements as outlined in this report, as well as a need to enable stakeholders to agree on critical areas that should be addressed. Some of the critical questions to be addressed are as follows:

- How far is far enough in expanding or going beyond the design basis?
- How should international input be obtained and coordinated in making improvements in the design basis?
- How is the design basis to be standardized and maintained on a worldwide fleet of plants given the goals of the on-going MDEP and WNA CORDEL efforts for new reactors?
- What are the appropriate risk metrics when using probabilistic methods for designing and operating reactors within the New Nuclear Safety Construct?
- What methods should be used for maintaining plant configuration for all the stations around the world, given the differences that exist among plant operator and reactor vendor engineering staffs with the critical skills for supporting the safety construct?
- How and to what degree can and should the INPO model be extrapolated and exported to increase global nuclear energy safety?
- How should regulators go about developing the technical basis for updating emergency planning zones worldwide?

- How should the global nuclear industry encourage development of a more-rational and better-understood approach to low-level radiation effects, including addressing whether use of the linear non-threshold approach is appropriate, based on latest available science worldwide?
- How should the industry apply the New Nuclear Safety Construct globally to operating plants and those under construction or being designed?
- How will the global nuclear industry establish the roles of the various stakeholders, including regulators, reactor vendors, operators and owners, national industry organizations, standards development organizations, and other international organizations such as NEA, WNA, and IAEA.

9.7.2 Updating and Expanding Nuclear Codes and Standards

The role of Codes and Standards is to independently promulgate rules and knowledge required for safe design, manufacturing, and operation of technological systems. The ASME is an accredited Standards Development Organization (SDO) that publishes and maintains codes and standards used in the nuclear power industry worldwide. Such standards include Rules for Construction of Nuclear Facility Components in Section III of the ASME Boiler and Pressure Vessel Code (BPVC), Rules for In Service Inspection of Nuclear Power Plant Components in Section XI of the ASME BPV Code, the ASME Code for Operation and Maintenance of Nuclear Power Plants, and the ASME/ANS Probabilistic Risk Assessment (PRA) Standard.

With regard to its commitment to incorporate new technology and industry knowledge into its nuclear standards, ASME has sought to collaborate with other SDOs in documenting the lessons learned from the events at Fukushima Dai-ichi and to develop revisions to standards that reflect those lessons learned. The potential need for new standards has also been recognized by others. For example, the ASME/ANS PRA Standard is prepared by the ASME/ANS Joint Committee on Nuclear Risk Management. This is the consensus standards committee responsible for PRA standards and for developing risk management codes and standards for risk-informed applications in the nuclear industry. This committee has generated a detailed strategic plan to develop and implement risk-management standards across all nuclear power plant operating modes. This committee should play a key role in implementation of an all-risk construct.

The ASME Board on Nuclear Codes and Standards (BNCS) has formed a Task Group on Design Basis and Response to Severe Accidents. The purpose of this Task Group is to manage collection and evaluation of data related to the Great East Japan Earthquake and Tsunami and the ensuing reactor accident at the Fukushima Dai-ichi station, and to recommend potential ASME codes and standards initiatives. The Task Group will share information with ANS, ASTM International, Institute of Electrical and Electronics Engineers, Inc. (IEEE), and other SDOs. It will also interact with other international SDOs and the IAEA for cooperative inputs, as appropriate. For example, the ASME BNCS Task Group has been reviewing work by the JSME on development of severe accident management guidelines.

The Task Group has initiated formation of several sub-tier task groups for technical evaluation of data. The scopes of these task groups are Design Basis External Events, Component Integrity, Safety System Response, and Severe Accident Mitigation and Response. These task groups will make recommendations to the appropriate ASME standards committees on potential code and standard development initiatives.

Among the areas of review are the following: treatment of design basis external event loadings and consequences; extended station blackout; hydrogen control; pressure boundary integrity; containment integrity; fuel pool integrity; and severe accident management guidance for response and recovery.

The Task Group will communicate and coordinate these initiatives with the NRC, the NEI and other U.S. and international industry stakeholders.

Recommended standards initiatives resulting from the work of the ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events will be integrated with the efforts of the ASME BNCS Task Group.

9.7.3 Dissemination of Conclusions and Guidance

ASME plans to disseminate and communicate the conclusions and guidance from this report and follow-on work that could include interactive public workshops, congressional briefings, visual and audio media, summary white papers, technical articles, and conferences. The communications will be developed in a manner consistent with a technically-sound energy portfolio, according to the needs and resources of different countries.

9.8 Summary Conclusions

The Fukushima Dai-chi accident has caused a global re-examination of the safety of nuclear power plants to severe external event challenges; the global response to the accident will have lasting effects on the operation of existing plants and the deployment of new ones, The ASME Task Force report has focused on the need to address socio-political and economic consequences of nuclear power plant accidents due to rare yet credible events, and on an approach to forging necessary elements for implementing a construct that delivers the requisite safety. The proposed approach will further improve protection of public health and safety and the environment.

The importance of society's response to natural events and to industrial accidents, regardless of cause, cannot be overemphasized. The large potential impact on society demands that industrial safety constructs be intrinsic to the protective actions and overall response framework, including addressing the vital role that communications play on the public interpretation of the safety and social repercussions of an accident. In the particular case of nuclear power, addressing accident consequences is not a final goal; the benefits of a new safety construct include improved socio-political and economic factors important to continuation and growth of nuclear power worldwide.

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APPENDIX A – THE SOCIAL IMPACT OF NUCLEAR POWER

A.1 Introduction

The ASME Task Force considers nuclear power to be an important part of the energy supply portfolio and an important option for the global energy mix—now and in the foreseeable future. The ASME Task Force believes that its recommendations will strengthen the infrastructures needed to ensure that nuclear power remains an available source of safe and emission-free power. This report, by its very origins, is critical of nuclear power incidents and accidents—focusing on deficiencies that caused or contributed to their consequences, as well as on the lessons learned from the reactor accidents and near misses from over 50 years of operating experience. The recommendations being made to enhance the safety of nuclear reactors and support availability of this technology to future generations are necessary to provide the appropriate framework for decision-makers. A proper balance within this context is difficult to achieve when addressing accidents; for this reason, Appendix A has been written to provide a more balanced discussion of nuclear power. There are many good attributes as well as opportunities for improvement with this technology; this appendix discusses "both sides of the coin."

A.2 Economic and Socio-Political Impact of Major Reactor Accidents

A.2.1 Impact of the Fukushima Dai-ichi Accident

Reliable estimates of the full economic impact of the Fukushima Dai-ichi accident are difficult to obtain. This is the first time such a major cleanup effort has been undertaken in an open, democratic nation. Furthermore, the reactor accident occurred in the midst of the Great East Japan Earthquake and Tsunami, which itself did substantial damage to the east coast region of Japan and its economy—beyond the effects at nuclear power plants—confounding the task of separating the cost impacts into those attributable to the reactor accident and the earthquake/tsunami. The ASME Presidential Task Force examined current cost estimates attributable to the Fukushima Dai-ichi accident in the following categories:

- *Replacement power costs due to shutdown of all 54 Japanese nuclear power plants after the Fukushima Dai-ichi accident, pending safety reviews (stress tests) and approval to restart:* Prior to the Fukushima Dai-ichi accident, nuclear power supplied approximately 30% of the electrical generation in Japan. World Nuclear News (Jan 25, 2012) reported that the resultant increase in fossil fuel imports, primarily Liquefied Natural Gas (LNG), to replace lost nuclear generating capacity cost $55 billion U.S. dollars through the final 9 months of 2011, and attributed this as a major factor in the county's overall trade deficit for 2011 of $32 billion U.S. dollars, the first such deficit posted in Japan since 1980.

- *Direct cleanup and decommissioning cost for Units 1-4 at the Fukushima Dai-ichi plant:* The Japanese government estimates that this could take 40 years to complete. In June 2011, the Japan Atomic Industrial Forum (JAIF) estimated this cleanup might cost upward of $250 billion U.S. dollars. According to JAIF (January 30, 2012), initial loan guarantees of the equivalent of $65 billion U.S. dollars were provided in the 2012 Japanese national budget for this purpose.

- *Environmental cleanup of radiologically contaminated lands outside the Fukushima Dai-ichi plant boundary:* Japanese Prime Minister Noda (as reported by Reuters on October 20, 2011) estimated a cost of $13 billion U.S. dollars to rehabilitate all of the contaminated lands. According to JAIF (January 30, 2012), the equivalent of $5.9 billion U.S. dollars has been earmarked in the 2012 Japanese national budget for this purpose. The actual cost of cleaning radiologically contaminated areas will depend on the final dose levels that are being used and will be used as acceptable for specific purposes, which will be a difficult and region-dependent decision.

- *Compensation for citizens evacuated from their homes during and after the accident:* JPMorgan (as reported by Reuters on April 11, 2011) initially estimated this cost at approximately $23 billion U.S. dollars, although this amount is highly dependent on the length of time people are restricted from returning to their homes and is subject to the outcome of inevitable legal proceedings. Bloomberg reported this cost as high as $58 billion U.S. dollars (January 12, 2012). According to JAIF (January 30, 2012), the Japanese government established the Nuclear Damage Liability Facilitation Fund to handle compensation payments, and provided an initial delivery bond of the equivalent of $65 billion U.S. dollars..
- *Lost capitalization of TEPCO.* Bloomberg, Reuters, and others have reported a drop in the stock price of TEPCO of 90% since the Fukushima Dai-ichi accident, primarily as a result of the uncertain financial liability to the company. This equates to a $30 billion U.S. dollar loss to the 1.3B shares of TEPCO stock, although some of this is expected to be recovered over time.
- *Lost commerce in the East Japan region as a result of the Fukushima Dai-ichi accident and radiological contamination:* Since this region contributes only about 2.5% to the Japanese GDP, this loss is expected to be small and is ignored in this analysis. However, as a result of general fear of potential contamination of food and other products from Japan, there is likely to be a much greater economic impact than can be directly inferred from the regional contribution to the Japanese GDP.

The estimated total cost of the Fukushima Dai-ichi accident is therefore currently approaching $500 billion U.S. dollars. This can only increase, in the future because of the additional imported LNG to replace the power from the shuttered nuclear power plants and from other losses of commerce. As of May 5 2012, all Japanese nuclear power plants are shut down for an undefined period while safety assessments and improvements are being made. The local prefectures have the final approval authority once the Government of Japan approves restart. It is not known at this time when or if the local prefectures will allow restart of the plants in their regions or if the Japanese government will decide to phase out nuclear power in the country. If Japan abandons nuclear power, there will be staggering economic and environmental impacts, including shutdown of much of the Japanese nuclear infrastructure and potential loss of nuclear power technology export business.

There are still tens of thousands of people displaced from their communities from the regions evacuated for radiological protection in Japan’s Eastern seaboard. These people continue to be stressfully impacted by the accident, and it is unclear when they will be allowed to return to their homes, and under what conditions. Compensation and care of these evacuees will continue during the cleanup operations.

Because the events at Fukushima Dai-ichi have posed no significant radiological hazards to the public, it is clear that the reactor accident resulted in only a socio-political, financial disaster. The magnitude of this disaster provides the motivation for forging a new nuclear safety construct.

A.2.2 Impact of the Chernobyl Accident

The Chernobyl reactor accident was the worst in the history of nuclear power. Because of the reactor type and because the reactor had no containment building, a considerable amount of radioactive material was released to the environment. In addition, notification and evacuation of the nearby public was delayed by the political structure of the government. The major consequences from the accident can be briefly summarized as follows [4]:

- Worldwide impact on nuclear power plant closures and deployment, especially in Europe, is estimated in the range of $250 billion to $500 billion U.S. dollars over 25 years.
- Major psychological and sociological impact on millions of people, with over 300,000 people resettled.
- Major disruption of land, habitat, and workplaces. The IAEA notes that about 150,000 square kilometers in Belarus, Russia, and Ukraine were contaminated, with an area within 30 kilometers (18 miles) around the plant declared an "exclusion zone."
- Presently, about 7 million people in the most affected regions still receive compensation or allowances related to their role in recovering from or being affected by the accident.

In the words of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), "The [Chernobyl] accident caused serious social and psychological disruption in the lives of those affected and vast economic losses over the entire region. Large areas of the three countries were contaminated with radioactive materials, and radionuclides from the Chernobyl release were measurable in all countries of the northern hemisphere … Among the residents of Belarus, the Russian Federation and Ukraine, **there had been up to the year 2005 more than 6,000 cases of thyroid cancer reported in children and adolescents** who were exposed at the time of the accident, and more cases can be expected during the next decades ... **Apart from this increase, there is no evidence of a major public health impact attributable to radiation exposure two decades after the accident.** There is no scientific evidence of increases in overall cancer incidence or mortality rates or in rates of non-malignant disorders that could be related to radiation exposure. The incidence of leukemia in the general population, one of the main concerns owing to the shorter time expected between exposure and its occurrence compared with solid cancers, does not appear to be elevated. Although those most highly exposed individuals are at an increased risk of radiation-associated effects, the great majority of the population is not likely to experience serious health consequences as a result of radiation from the Chernobyl accident. Many other health problems have been noted in the populations that are not related to radiation exposure." [92]

It is now accepted that 15 young people died from complications from thyroid cancer, which can be attributed to the Chernobyl reactor accident. It is an established fact that 48 workers or firemen who were severely exposed while trying to terminate the reactor fire, and suffered acute radiation syndrome, died of their injuries within 10 years of the accident.

A.2.3 Impact of the Three Mile Island 2 Accident

TheTMI-2 accident presented no measurable health risk to the public or plant staff. Studies conducted by the NRC, the EPA, the Department of Health, Education and Welfare, the DOE, and the State of Pennsylvania have examined the radiological health consequences of the releases following the accident. The average dose to the 2 million people living in the area is estimated to be 1 millirem (0.01 millisievert) in addition to the ~ 100 millirem (~ 1 millisievert) this population normally receives in any given year from naturally-occurring sources of radiation. These and other independent studies led to the conclusion contained on the NRC website, "Backgrounder on the Three Mile Island Accident," which states, "in spite of serious damage to the reactor, most of the radiation was contained and … the actual release had negligible effects on the physical health of individuals or the environment." The Presidential Commission on Three Mile Island reported, "…there will be no cases of cancer or the number of cases will be so small that it will never be possible to detect them. The same conclusion applies to the other possible health effects." [93]

In addition to the total loss of the TMI-2 unit valued at several billion dollars, the economic and socio-political impact of the TMI accident included the following:

- Cleanup of the reactor accident, which started in August 1979 and ended in December 1993, at a total cost of approximately $1 billion U.S. dollars
- Loss of $900 million U.S. dollars per year to the Pennsylvania economy
- Backfit and other costs for the nuclear fleet of reactors in the U.S. were estimated at $10-60 billion U.S. dollars
- Evacuation of 140,000 pregnant women and pre-school children from the area for a short time
- Crystallization of anti-nuclear safety concerns among non-governmental organizations and some of the general public, which has persisted for decades
- Cancellation of many new reactors already under construction and undermining of confidence in the nuclear industry, associated with a level of risk not aligned with the actual consequences of the accident.

A.3 The Present and Future Value of Nuclear Power in the U.S.

There are currently 104 operating reactors in the U.S., providing 20% of the electrical generation. The Energy Information Administration Annual Energy Outlook 2011 [94] projects 31% increase in U.S. electricity consumption by 2035.

The fleet of U.S. reactors today represents an asset value of $100 to $200 billion U.S. dollars. These reactors, which can continue to provide reliable power generation for another 20 to 40 years, depending on plant age and life extension already approved. Figure A-1 depicts U.S. nuclear generation relative to plant lifetimes for 40-year licenses, extended licensed life from 40 to 60 years, and for extending life beyond 60 years. Virtually all of the existing U.S. commercial nuclear reactors either already have extended or will be extending the licensed life of plants for another 20 years. Extending lifetimes beyond 60 years presents challenging life-limiting issues as discussed in an ASME Long Term Operations Workshop in 2010. [95]

The red line in Figure A-1 represents the nuclear electrical generating capacity required to maintain the current 20% share of nuclear power in meeting the growing electricity demand in the U.S. To meet this conservative demand, almost twice as many nuclear plants will be needed in the year 2050 as are in operation today. Unless long term operation beyond 60-year plant lifetimes is achieved, the U.S. will need to construct at least 5 large nuclear power plant units per year. This can be compared with the situation today in which a total of only five units are under construction (the long delayed Watts Bar Unit 2 and the four recently-approved Combined Construction and Operating Licenses for four new units in Georgia and South Carolina). With nuclear power currently providing the vast majority of the total emission-free electrical generation in the U.S., achieving this level of emission-free base-load electrical generation growth by other reliable sources will be difficult, if not impossible. A similar situation will exist for many other countries—given the projected worldwide growth in electricity demand—particularly in Asia.

The key point is that safety in nuclear power generation will improve, as with all technologies, as a result of the lessons-learned from previous accidents and events. Improvements in design, plant operations, and human performance incorporated into the U.S. nuclear fleet following the TMI-2 accident resulted in about an order of magnitude in risk reduction in the likelihood of a severe accident with significant radioactivity release. It is to be expected that the improvements suggested and encouraged in and by the New Nuclear Safety Construct could result in a similar risk reduction.

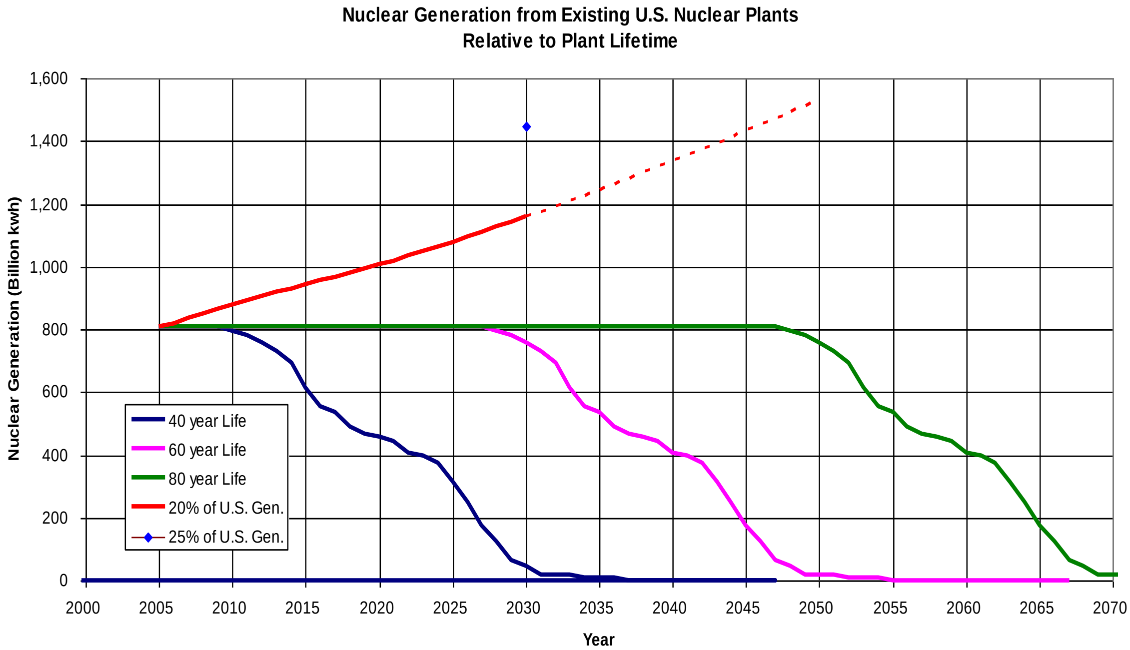

Figure A-1 – Nuclear Generation from Existing U.S. Nuclear Plants

A.4 The Reliability of Nuclear Power

There are a number of benefits derived from normal operation of nuclear power plants. They provide a reliable and safe source of emission-free base-load electricity that helps reduce reliance on fossil fuel alternatives. As shown in Chapter 2 (Figure 4), the fleet-wide performance of U.S. nuclear plants over the 1980s and 1990s has improved dramatically in terms of both capacity factors and significant safety events, such that nuclear power has been the most reliable source of power in the U.S. over the past decade. A comparison of the capacity factors of other power generation sources in the U.S. is shown in Table A-1. It is acknowledged that utility or dispatcher decisions on which plants to bring on line or at what power they operate can change the operating capacity factors of different types of plants. Traditionally, nuclear plants have been run as base-load units, as have large coal plants. On the other hand, natural-gas combined-cycle plants have typically been used to follow load demands. Such decisions are based on fuel prices, operating costs, availability of variable renewable energy sources, responsiveness of technologies to load follow, emissions, etc. With the current price of pipeline natural gas being low in the U.S., it is likely that combined-cycle gas plants will assume a greater role as base-load units. Therefore, it is expected their capacity factors will increase in the future and capacity factors for coal plants will decrease—particularly for the older, smaller coal plants, which tend to have lower efficiencies and higher greenhouse gas emissions.

Table A-1 – Average Capacity Factor in the U.S. by Energy Source (1998 – 2009)

Year	Coal	Petroleum	Natural Gas CC[1]	Natural Gas Other	Nuclear	Hydroelectric Conventional	Other Renewables	All Energy Sources
1998	67.7	22.2	--	34.2	79.2	46.6	57.0	54.6
1999	68.1	22.4	--	33.2	85.3	45.9	56.9	54.9
2000	71.0	20.5	--	37.1	87.7	39.5	59.1	54.6
2001	69.2	21.5	--	35.7	89.4	31.4	50.2	51.4
2002	70.0	18.1	--	38.2	90.3	38.0	54.0	49.7
2003	72.0	22.4	33.5	12.1	87.9	40.0	50.0	47.7
2004	71.9	23.3	35.5	10.7	90.1	39.4	50.5	47.9
2005	73.3	23.8	36.8	10.6	89.3	39.8	47.0	48.3
2006	72.6	12.6	38.8	10.7	89.6	42.4	45.7	48.0
2007	73.6	13.4	42.0	11.4	91.8	36.3	40.0	48.7
2008	72.2	9.2	40.6	10.6	91.1	37.2	37.3	47.4
2009	63.8	7.8	42.2	10.1	90.3	39.8	33.9	44.9

[1] Prior to 2003, the generation collected on Form EIA-906 did not have a distinction for combined cycle (CC) prime movers. All natural gas-fired plants of all types are included in "Natural Gas Other" for 1998 to 2002.
Note: Technical Note: Average Capacity Factor is the ratio of actual generation to maximum potential output, expressed as a percent.
Average Capacity Factor = [(Net Generation)/(Net Summer Capacity* Number of Hours in the Year)] * 100
for the respective energy source and yea
Sources: U.S. Energy Information Administration, Form EIA-860, "Annual Electric Generator Report;" Form EIA-923, Power Plant Operations Report," and predecessor forms.

It is clear from these capacity factor values that the 104 operating nuclear power reactors, which provide approximately 20% of U.S. electricity, are valuable assets to the U.S. economy.

A.5 Price Stability and Energy Security

Because of the high energy content of nuclear fuel and the small percentage cost of the fuel (~15%) to the total power-generation cost, use of nuclear power provides stability and a high degree of certainty to the cost of electricity. Uranium fuel is easy to stockpile, because of its relatively-small volume and high energy content; the resulting cost of electricity from nuclear power is therefore very insensitive to fluctuations in the price of uranium. This provides a good degree of energy security and price stability to the base load power market because nuclear power is not subject to price volatility. Figure A-2 shows the historical trend in the production cost (fuel plus operations and maintenance costs) of the major thermal power generation sources; the costs shown do not include the initial capital costs, which are highest for nuclear power plants. Nuclear power provides the lowest production cost, followed closely by coal, then natural gas, and finally petroleum. Most important of these factors is the stability of costs shown for coal and nuclear generation.

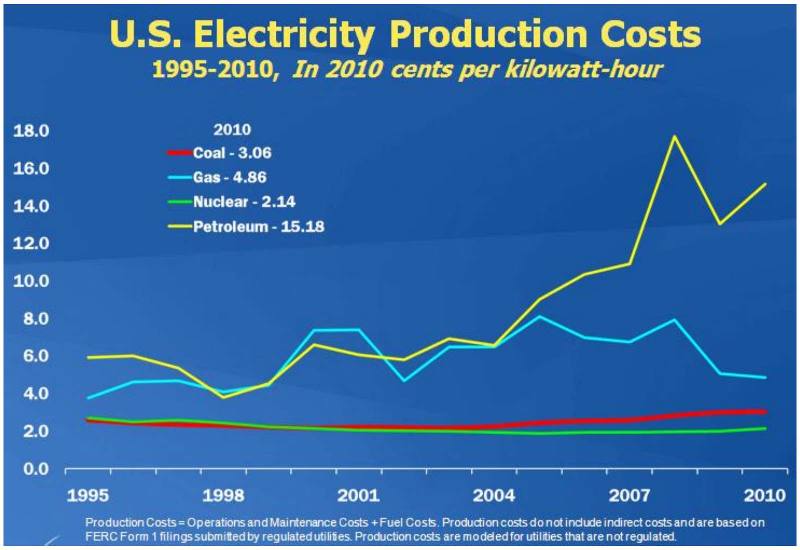

Figure A-2 – U.S. Electricity Production Costs by Fuel Type

A.6 Greenhouse Gas Emissions

There is strong support in the U.S. and worldwide for moving toward lower-carbon-emission technologies for power generation. In the U.S., more than 50% of electricity is generated by coal-fired plants, which emit large amounts of CO_2. The worldwide trend shows even a greater reliance on coal followed by natural gas and oil (see Figure A-3). This general trend is projected to continue because of the availability of fossil fuels and their affordability compared to other energy sources. However, with increased concern for the global impact of greenhouse gas emissions, this trend could change and nuclear power could be one of the major contributors to emission-free energy.

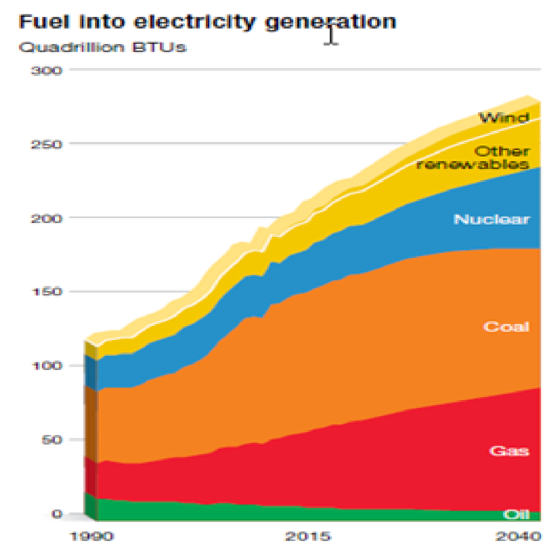

Figure A-3 – Power Generation by Fuel Type (1990 to 2040)

The best measure of the contribution to CO_2 emissions from different fuel types used to generate power is the life cycle emissions. This includes the CO_2 produced to build the power plant, to provide the fuel, to operate the plant, and finally to decommission the plant at the end of its life. Nuclear power is one of the lowest emitters. [96]

As a benchmark, over the past year—with the shutdown of all its nuclear power plants—Japan was forced to depend much more on fossil fuels.

- LNG imports hit an all-time high in January 2012 and jumped 28% during 2011.
- Coal imports for electricity increased more than 26% in January 2012 from a year earlier and increased nearly 8% in 2011.
- Imports of crude oil increased 350% in March 2012 compared to March of last year; overall consumption increased 9% over the year earlier. [97]

This increased fossil fuel use will have a negative impact on the environment from increased greenhouse gas emissions.

At the present time, nuclear power plants provide nearly 69% of the emission-free electricity in the U.S., followed by hydroelectric power (~21%) and renewable power sources (~10%).

A.7 Relative Health Risks of Electrical Generation Sources

All energy technologies have risks to public health and safety as a result of accidents along the energy-delivery chains as well as normal operations. The stages of the energy-delivery chains include exploration and production, transportation, processing and storage, regional distribution, energy production, and waste treatment or disposal. Fatalities can occur at any of these stages, and different technologies have the majority of their fatalities associated with one or two of these stages. For example, coal risks are dominated by mine explosions, fires, and cave-ins. Oil risks are dominated by transportation and distribution accidents. Estimates of nuclear power fatalities, although relatively small, are dominated by the production stage—exclusively from very few reactor accidents. These fatalities include prompt fatalities from high radiation doses, as well as latent or long term health effects, i.e., radiation-induced cancer fatalities.

It is important to recognize that latent fatalities are an important consideration in properly accounting for the health effects due to normal operation, as well as from accidents associated with different energy sources. In fact, the estimated latent effects dominate risk profiles for nuclear generation, due to the historically-small contribution from prompt fatalities. The best scientific projections on latent cancer fatalities from UNSCEAR are far less than those previously estimated, and the case has been made that any fatalities from reactor accidents, e.g., Chernobyl-related cancer cases, will likely not be statistically distinguishable from the normally-occurring cancer rate for the affected regions.

These data demonstrate that nuclear power generation has the smallest contribution to risk to the general public of the electricity generators, even when the estimated severe-accident consequences are considered. It has the lowest estimated fatality rate per TWh according to currently published data. While current data predates the event at Fukushima Dai-ichi, there were no prompt fatalities nor are there expected to be significant latent health effects from this reactor accident. Therefore, the data are still relevant and support continued and expanded use of nuclear power technology. [98]

In summary, there are demonstrable and consistent benefits from nuclear power generation that should be considered in decisions effecting the continuation and growth of global nuclear power.

ACKNOWLEDGEMENTS

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ABBREVIATIONS AND ACRONYMS

10 CFR – U.S. Code of Federal Regulations, Title 10

9/11 – September 11, 2001

AC – (electrical) Alternating Current

ACRS – Advisory Committee on Reactor Safeguards

AEC – Atomic Energy Commission

ANPR – Advance Notice of Proposed Rulemaking

ANS – American Nuclear Society

AOP – Abnormal Operating Procedure

ASME – American Society of Mechanical Engineers

ASME Task Force – ASME Presidential Task Force on Response to Japan Nuclear Power Events

BBC – British Broadcasting Company

BNCS – Board on Nuclear Codes and Standards

BNL – Brookhaven National Laboratory

BPVC – ASME Boiler and Pressure Vessel Code

BWR – Boiling Water Reactor

CBT – Computer Based Training

Commission – U.S. Nuclear Regulatory Commission

CORDEL – Cooperation in Reactor Design Evaluation and Licensing

CRMP – Configuration Risk Management Program

DBA – Design Basis Accident

DC – (electrical) Direct Current

DCER – Design Certification Environmental Report

DOE – U.S. Department of Energy

EAL – Emergency Action Level

ECCS – Emergency Core Cooling Systems

EDMG – Extensive Damage Mitigation Guideline

EIS – Environmental Impact Statement

EOF – Emergency Operations Facility

EOP – Emergency Operating Procedure

EP – Emergency Preparedness

EPA – U.S. Environmental Protection Agency

EPRI – Electrical Power Research Institute

EPZ – Emergency Planning Zone

ERO – Emergency Response Organization

FEMA – Federal Emergency Management Agency

GDP – Gross Domestic Product

HPS – Health Physics Society

IAEA – International Atomic Energy Agency

ICRP – International Committee on Radiation Protection

IEEE – Institute of Electrical and Electronics Engineers, Inc.

INPO – Institute of Nuclear Power Operations

INSAG – International Nuclear Safety Group

IPE – Individual Plant Examination

JAIF – Japan Atomic Industrial Forum

JSME – Japan Society of Mechanical Engineers

LNG – Liquefied Natural Gas

LNT – Linear, Non-Threshold

LOCA – Loss-Of-Coolant Accident

LWR – Light Water Reactor

MDEP – Multinational Design Evaluation Program

NEA – Nuclear Energy Agency

NEI – Nuclear Energy Institute

NRC – U.S. Nuclear Regulatory Commission

NUREG – U.S. NRC Regulatory Guide

OECD – Organization for Economic Cooperation and Development

PAG – Protective Action Guide

PORV – Pilot Operated Relief Valve

PRA – Probabilistic Risk Assessment

PWR – Pressurized Water Reactor

RBMK – Reaktor Bolshoy Moshchnosti Kanalniy

ROP – Reactor Oversight Process

SAMA – Severe Accident Mitigation Alternatives

SAMDA – Severe Accident Mitigation Design Alternatives

SAMG – Severe Accident Management Guideline

SAR – Safety Analysis Report

SBO – Station Blackout

SDO – Standards Development Organization

SECY – Policy Paper Issued by the Secretary of the NRC

SMR – Small Modular Reactor

SOARCA – State-of-the-Art Reactor Consequence Analysis

TEDE – Total Effective Dose Equivalent

TMI-2 – Three Mile Island Unit 2

TSC – Technical Support Center

UNSCEAR – United Nations Scientific Committee on the Effects of Atomic Radiation

UK – United Kingdom

U.S. – United States

USSR – Union of Soviet Socialist Republics

WANO – World Association of Nuclear Operators

WNA – World Nuclear Association

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