LESSONS LEARNED FROM THE Fukushima Nuclear Accident FOR IMPROVING SAFETY AND SECURITY OF U.S. Nuclear Plants

PHASE 2

Committee on Lessons Learned from the Fukushima Nuclear Accident
for Improving Safety and Security of U.S. Nuclear Plants

Nuclear and Radiation Studies Board

Division on Earth and Life Studies

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Cover images: (Front) Satellite image of the Fukushima Daiichi Nuclear Power Station taken on March 14, 2011, while the accident was still in progress. The Unit 1 reactor building (lower-left corner of image) was damaged by a hydrogen explosion at 15:36 (3:36 P.M.) Japan Standard Time on March 12. The Unit 3 reactor building (building with steam plume near center of image) was damaged by a hydrogen explosion at 11:01 the day the image was recorded. The Unit 4 reactor building, which is still intact in this image, would be damaged by a hydrogen explosion at 06:14 the next day (March 15). (Back) Satellite image of the harbor-front of the Fukushima Daiichi Nuclear Power Station showing extensive damage from the tsunami. Also shown is a seismogram of the Great East Japan Earthquake recorded at the Oshika K-Net Station (MYG011) in Miyagi Prefecture. Satellite image courtesy of DigitalGlobe©2014.

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This report has been reviewed in draft form by individuals chosen for their diverse perspectives and technical expertise in accordance with procedures approved by the Report Review Committee of the National Research Council. The purpose of this independent review is to provide candid and critical comments that will assist the National Research Council in making this published report as sound as possible and will ensure that this report meets institutional standards for objectivity, evidence, and responsiveness to the study charge. The review comments and draft manuscript remain confidential to protect the integrity of the deliberative process. We wish to thank the following individuals for their participation in the review of this report:

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Although the reviewers listed above provided many constructive comments and suggestions, they were not asked to endorse the conclusions and recommendations of this report, nor did they see the final draft of the report before its release. The review of this report was overseen by Marvin L. Adams, Texas A&M University, and Ali Mosleh, University of California, Los Angeles. Appointed by the National Research Council, Drs. Adams and Mosleh were responsible for making certain that an independent examination of this report was carried out in accordance with institutional procedures and that all review comments were considered carefully. Responsibility for the final content of this report rests entirely with the authoring committee and the institution.

Contents

Synopsis

The U.S. Congress asked the National Academy of Sciences[1] to examine the causes of the March 11, 2011, accident at the Fukushima Daiichi nuclear plant and identify lessons learned for the United States. NRC (2014) contains the first part of this examination; the present report, which contains the second and final part, focuses on three issues: (1) lessons learned from the accident for nuclear plant security, (2) lessons learned for spent fuel storage, and (3) reevaluation of conclusions from previous Academies studies on spent fuel storage. Brief descriptions of key selected recommendations are provided in this Synopsis. Additional details are provided in the Summary and individual chapters.

Lessons Learned from the Accident for Nuclear Plant Security. Nuclear plant operators and their regulators should upgrade and/or protect nuclear plant security infrastructure and systems and train security personnel to cope with extreme external events and severe accidents. Such upgrades should include independent, redundant, and protected power sources dedicated to plant security systems that will continue to function independently if safety systems are damaged; diverse and flexible approaches for coping with and reconstituting plant security infrastructure, systems, and staffing during and following extreme external events and severe accidents; and training of security personnel on the use of these approaches.

[1] Now the National Academies of Sciences, Engineering, and Medicine.

Lessons Learned from the Fukushima Daiichi Accident for Spent Fuel Storage. The U.S. nuclear industry and its regulator should give additional attention to improving the ability of plant operators to measure real-time conditions in spent fuel pools and maintain adequate cooling of stored spent fuel during severe accidents and terrorist attacks. These improvements should include hardened and redundant physical surveillance systems (e.g., cameras), radiation monitors, pool temperature monitors, pool water-level monitors, and means to deliver pool makeup water or sprays even when physical access to the pools is limited by facility damage or high radiation levels.

Reevaluation of Conclusions from Previous National Academy of Sciences Studies on Spent Fuel Storage. The present report provides a reevaluation of the findings and recommendations from NRC (2004, 2006). Two key recommendations emerged from this reevaluation concerning the application of risk assessment to security applications: (1) the U.S. nuclear industry and the U.S. Nuclear Regulatory Commission should strengthen their capabilities for identifying, evaluating, and managing the risks from terrorist attacks and (2) the U.S. Nuclear Regulatory Commission should sponsor a spent fuel storage security risk assessment for U.S. nuclear plants.

The Academies also examined the U.S. Nuclear Regulatory Commission's Spent Fuel Pool Study (USNRC, 2014a) and Expedited Transfer Regulatory Analysis (USNRC, 2013) to assess their responsiveness to the recommendations in NRC (2004, 2006). One recommendation emerged from this examination: the U.S. Nuclear Regulatory Commission should perform a spent fuel storage risk assessment to elucidate the risks and potential benefits of expedited transfer of spent fuel from pools to dry casks. This risk assessment should address accident and sabotage risks for both pool and dry storage.

Summary

The U.S. Congress asked the National Academy of Sciences[1] (NAS) to conduct a technical study on lessons learned from the Fukushima Daiichi nuclear accident for improving safety and security of commercial nuclear power plants[2] in the United States. The complete study task is given in Sidebar 1.2 in Chapter 1.

This study was carried out in two phases: Phase 1 focused on the causes of the Fukushima Daiichi accident and safety-related lessons learned for improving nuclear plant systems, operations, and regulations exclusive of spent fuel storage. The phase 1 report was issued in 2014 (NRC, 2014).[3]

Phase 2 (this study) focused on three tasks:

1. Security-related lessons learned from the Fukushima Daiichi accident for improving nuclear plant systems, operations, and regulations;
2. Lessons learned from the accident for improving safety of spent fuel storage; and
3. Reevaluation of the findings and recommendations from previous NAS reports on spent fuel storage safety and security.

The Academies committee that carried out this study (hereafter referred to as the *committee*) provides findings and recommendations to address

[1] Now the National Academies of Sciences, Engineering, and Medicine.
[2] The terms *nuclear power plant* and *nuclear plant* are used interchangeably in this report.
[3] The report is available at http://www.nap.edu/catalog/18294/lessons-learned-from-the-fukushima-nuclear-accident-for-improving-safety-of-us-nuclear-plants.

these study tasks in Chapters 2, 3, and 4 of this report. Summarized versions of selected key findings and recommendations are presented in the following sections.

TASK 1: SECURITY-RELATED LESSONS LEARNED FOR PLANT SYSTEMS, OPERATIONS, AND REGULATIONS[4]

The March 11, 2011, Great East Japan Earthquake and tsunami caused extensive damage to safety and security infrastructure at the Fukushima Daiichi plant. Tsunami damage and power losses affected the integrity and operation of numerous security systems, including lighting, physical barriers and other access controls, intrusion detection and assessment equipment, and communications equipment. Workers monitoring the protected area of the plant evacuated to higher ground just before the tsunami arrived at the plant, and some security workers were temporarily evacuated from the plant on the fourth day of the accident.

The committee finds (**Finding 3.1**) that extreme external events and severe accidents can cause widespread and long-lasting (i.e., days to weeks) disruptions to security infrastructure, systems, and staffing at nuclear plants that can create opportunities for malevolent acts and increase the susceptibility of critical plant systems to these acts. The committee recommends (**Recommendation 3.1**) that nuclear plant operators and their regulators upgrade and/or protect nuclear plant security infrastructure and systems and train security personnel to cope with extreme external events and severe accidents. The committee judges that the following three actions are needed:

1. Ensuring that there is adequate separation of plant safety and security systems so that security systems can continue to function independently if safety systems are damaged. In particular, security systems need to have independent, redundant, and protected power sources;
2. Implementing diverse and flexible approaches for coping with and reconstituting plant security infrastructure, systems, and staffing during and following external events and severe accidents; and
3. Training of security personnel on implementing approaches for reconstituting security infrastructure and systems.

The committee sees an opportunity for the nuclear industry to expand its FLEX initiative[5] to include critical security-related equipment, such as

[4] See Chapter 3 of this report for additional discussion of this task.

[5] Diverse and Flexible Coping Strategies to maintain/restore reactor and spent fuel pool cooling and reactor containment function.

access control, intrusion detection, and assessment, communications, and portable-lighting equipment. This equipment would need to be sufficiently standardized so that it could be used across the U.S. nuclear plant fleet and protected against extreme external events, severe accidents, and sabotage. Security personnel at U.S. plants would need to be trained on the use of this equipment if it were different from existing equipment at their plants.

TASK 2: LESSONS LEARNED FOR SPENT FUEL STORAGE[6]

Spent fuel was stored in eight locations at the Fukushima Daiichi plant on March 11, 2011: in spent fuel pools in each of the six reactor units (Units 1-6), in a common spent fuel pool, and in a dry cask storage facility. The present report focuses on spent fuel storage in the Unit 1-4 pools because these units sustained severe damage as a result of the March 11, 2011, earthquake and tsunami.

The committee finds (**Finding 2.1**) that the spent fuel storage facilities (pools and dry casks) at the Fukushima Daiichi plant maintained their containment functions during and after the March 11, 2011, earthquake and tsunami. However, the loss of power, spent fuel pool cooling systems, and water level- and temperature-monitoring instrumentation in Units 1-4 and hydrogen explosions in Units 1, 3, and 4 hindered efforts by plant operators to monitor conditions in the pools and restore critical pool-cooling functions. Plant operators had not planned for or been trained to respond to the conditions that existed in the Unit 1-4 spent fuel pools after the earthquake and tsunami. Nevertheless, they successfully improvised ways to monitor and cool the pools using helicopters, fire trucks, water cannons, concrete pump trucks, and ad hoc connections to installed cooling systems. These improvised actions were essential for preventing damage to the stored spent fuel and the consequent release of radioactive materials to the environment. The committee recommends (**Recommendation 2.1**) that the U.S. nuclear industry and its regulator give additional attention (described in Chapter 2) to improving the ability of plant operators to monitor real-time conditions in spent fuel pools and maintain adequate cooling of stored spent fuel during severe accidents and terrorist attacks.

The spent fuel pool in Unit 4 was of particular concern because it had a high decay-heat load. The committee used a steady-state energy-balance model to provide insights on water levels in the Unit 4 pool during the first 2 months of the accident (i.e., between March 11 and May 12, 2011). This model suggests that water levels in the Unit 4 pool declined to less than 2 m (about 6 ft) above the tops of the spent fuel racks by mid-April 2011. The model also suggests that pool water levels would have dropped

[6] See Chapter 2 of this report for additional discussion of this task.

below the top of active fuel[7] had there not been leakage of water into the pool from the reactor well and dryer/separator pit through the separating gates. This water leakage was accidental; it was also fortuitous because it likely prevented pool water levels from reaching the tops of the fuel racks. The events in the Unit 4 pool show that gate leakage can be an important pathway for water addition or loss from some spent fuel pools and that reactor outage configuration can affect pool storage risks.

The events in Unit 4 pool have important implications for accident response actions. As water levels decrease below about 1 m above the top of the fuel racks, radiation levels on the refueling deck and surrounding areas will increase substantially, limiting personnel access. Moreover, once water levels reach approximately 50 percent of the fuel assembly height, the tops of the rods will begin to degrade, changing the fuel geometry and increasing the potential for large radioactive material releases into the environment.

TASK 3: REEVALUATION OF THE FINDINGS AND RECOMMENDATIONS FROM PREVIOUS NAS REPORTS[8]

The "previous NAS reports" referred to in Charge 2 of this study (see Sidebar 1.2 in Chapter 1) refer to a single study carried out in 2003-2004 at the request of the U.S. Congress. That study produced two reports: a report containing classified and other security-sensitive information, hereafter referred to as the classified report (NRC, 2004), and an abbreviated version of this classified report that was suitable for unrestricted public release, hereafter referred to as the public report (NRC, 2006). The public report is similar in content to the classified report and contains all of its findings and recommendations. However, redactions and wording modifications were made to the classified report, including its findings and recommendations, to remove classified and other security-sensitive information.

Table 4.1 in Chapter 4 (pp. 102-110) summarizes the committee's reevaluation of the findings and recommendations in the public report (NRC, 2006). The left column of the table displays the findings and recommendations in NRC (2006) organized by their order of presentation in that report. The committee's reevaluation is presented in the right column of the table, also in the form of findings and recommendations. Selected key findings and recommendations from Table 4.1 are described below.

[7] The tops of the spent fuel racks are designed to be slightly taller than the top of active fuel.
[8] See Chapters 4-7 of this report for additional discussion of this task.

Terrorist Attacks on Spent Fuel Storage or Theft of Spent Fuel

NRC (2006) concluded that the terrorist attack risks could not be addressed "using quantitative and comparative risk assessments." Instead, the report examined "a range of possible terrorist attack scenarios in terms of (1) their potential for damaging spent fuel pools and dry storage casks; and (2) their potential for radioactive material releases" (NRC, 2006, p. 27).

The present committee agrees with NRC (2006) that there are technical challenges associated with identifying terrorist attack scenarios and quantifying their likelihoods. However, the committee judges that the NRC (2006) report's focus on quantification is too narrow a perspective for judging the feasibility of applying risk assessment methods to nuclear plant security. The committee finds (**Finding 4.1** in Table 4.1) that understanding of security risks at nuclear power plants and spent fuel storage facilities can be improved through risk assessment.

Risk assessment can help to broaden scenario identification, including cyber and asymmetric attack[9] scenarios; account for the performance of plant security personnel in responding to the identified scenarios; identify potential onsite and offsite consequences of such scenarios, ranging from radioactive releases to psychological impacts; and better characterize uncertainties. The identification of scenarios may be incomplete and the estimates developed through expert elicitation are subjective and can have large uncertainties. Nevertheless, risk assessment methods can provide useful security insights.

The committee recommends (**Recommendation 4.1A**) that the U.S. nuclear industry and the U.S. Nuclear Regulatory Commission (USNRC) strengthen their capabilities for identifying, evaluating, and managing the risks from terrorist attacks. The committee also recommends (**Recommendation 4.1B**) that the USNRC sponsor a spent fuel storage security risk assessment of sufficient scope and depth to explore the benefits of this methodology for enhancing security at U.S. nuclear plants.

NRC (2006) recommended that the USNRC obtain an independent review of surveillance and security measures for protecting stored spent fuel. The committee finds (**Finding 4.3**) that the USNRC has not obtained this review. The committee recommends (**Recommendation 4.3**) that this independent review include an examination of the effectiveness of the USNRC's security and surveillance measures for addressing the *insider*[10] *threat*. This threat can also be addressed using the committee-recommended security risk assessment (**Recommendation 4.1B**).

[9] See Section 3.2.1 in Chapter 3 for a discussion of asymmetric threats.

[10] An *insider* is a person who is authorized to have physical and/or cyber access to nuclear plant facilities and systems and is working alone or with outsiders to attack the plant.

Safety and Security of Pool Storage

The safe storage of spent fuel in pools depends critically on the ability of nuclear plant operators to keep the stored fuel covered with water. This fact was understood more than 40 years ago and was powerfully reinforced by the Fukushima Daiichi accident. If pool water is lost through an accident or terrorist attack,[11] then the stored fuel can become uncovered, possibly leading to fuel damage including runaway oxidation of the fuel cladding (a *zirconium cladding fire*) and the release of radioactive materials to the environment. NRC (2006) reviewed work that was being carried out by the USNRC and others to better understand how stored fuel can become uncovered as well as the consequences of such exposure.

NRC (2006) identified three measures that appear to have particular merit for reducing the likelihood of zirconium cladding fires following loss-of-pool-coolant events:

1. Developing a redundant and diverse response system to mitigate loss-of-pool-coolant events that would be capable of operation even if the pool or overlying building were severely damaged;
2. Reconfiguring spent fuel in the pools (i.e., redistribution of high-decay-heat assemblies so that they are surrounded by low-decay-heat assemblies) to more evenly distribute decay-heat loads and enhance radiative heat transfer; and
3. Limiting the frequency of offloads of full reactor cores into spent fuel pools, requiring longer shutdowns of the reactor before any fuel is offloaded, and providing enhanced security when such offloads must be made.

The committee received briefings and technical reports from USNRC and its contractor Sandia National Laboratories on additional technical analyses and physical experiments that have been carried out since NRC (2006) was released. The committee finds (**Finding 4.5**) that these USNRC and Sandia technical analyses confirm that reconfiguring spent fuel in pools can be an effective strategy for reducing the likelihood of fuel damage and zirconium cladding fires following loss-of-pool-coolant events. If a loss-of-coolant event results in fuel exposure, then reconfiguration may provide additional time for mitigating actions to be taken. However, reconfiguring the fuel does not eliminate the risks of zirconium cladding fires in all cases.

The USNRC and Sandia National Laboratories have performed physical experiments and computer analysis using the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code to analyze

[11] Such occurrences are referred to as *loss-of-pool-coolant events*.

loss-of-coolant events in spent fuel pools. These studies examined whether zirconium cladding fires could develop in the stored fuel assemblies and propagate to other assemblies in the pool; whether specific configurations of fuel in the pool could delay or prevent these fires from occurring; and whether certain mitigating strategies are effective for preventing this from occurring. The committee finds (**Finding 4.6**) that this additional work has substantially improved the state of knowledge concerning spent fuel behavior following partial or complete loss of pool water. However, experimental validation of the codes has not been performed for partially drained pools. The committee recommends (**Recommendation 4.6**) that the USNRC sponsor an end-to-end validation of the MELCOR code for modeling loss of coolant in spent fuel pools and validate key submodels.

The committee also finds (**Finding 4.7**) that the USNRC has not analyzed the potential vulnerabilities of spent fuel pools to the specific terrorist attack scenarios identified in NRC (2004). The agency has made good progress in implementing actions recommended in NRC (2006) to reduce the consequences of zirconium cladding fires (**Finding 4.8**). The committee recommends (**Recommendation 4.8**) that the USNRC and industry take additional steps to improve capabilities for further reducing and mitigating the risks of zirconium cladding fires. These steps are described in Chapter 6.

Safety and Security of Dry Cask Storage and Comparison with Pool Storage

The USNRC is performing additional analysis on dry cask vulnerabilities and incorporating results into its regulations through rulemaking. The vulnerability studies are addressing a range of attack scenarios and appear to be well conceived. However, because this work was still under way when the present study was being completed, the committee finds (**Finding 4.9**) that it is unable to assess that work's technical soundness and completeness. At the time the present report was being written, the USNRC's Independent Spent Fuel Storage Installation security rulemaking actions had not been completed and its future was not certain.[12] Consequently, the committee also finds (**Finding 4.10**) that it is unable to evaluate the technical soundness and completeness of this rulemaking. The committee recommends (**Recommendation 4.10**) that the USNRC should give high priority to completing these analyses and rulemaking.

The USNRC has completed technical and regulatory studies[13] to inform

[12] On October 6, 2015, the Commission approved a 5-year delay in the commencement of this rulemaking. See http://pbadupws.nrc.gov/docs/ML1528/ML15280A105.pdf.

[13] These studies are referred to as the Spent Fuel Pool Study (USNRC, 2014a) and Expedited Transfer Regulatory Analysis (USNRC, 2013). See Chapter 7 of this report.

a regulatory decision on the need for earlier-than-planned movements (*expedited transfer*) of spent fuel at commercial nuclear plants from pools to dry cask storage. These USNRC studies are valuable technical contributions to understanding the consequences of spent fuel pool accidents. However, the present committee finds (**Finding 4.11**) that these analyses did not consider spent fuel storage sabotage risks, dry cask storage risks, or certain health consequences that would likely result from a severe nuclear accident. The analysis also used simplifying bounding assumptions that make it technically difficult to assign confidence intervals to the consequence estimates or make valid risk comparisons. The committee recommends (**Recommendation 4.11**) that the USNRC perform a spent fuel storage risk assessment to elucidate the risks and potential benefits of expedited transfer of spent fuel from pools to dry casks. This risk assessment should address accident and sabotage risks for both pool and dry storage. The committee judges that this analysis is needed to address Finding 4E in NRC (2006) on whether "earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants."

The committee's critiques of the Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis are intended to strengthen the quality of future technical analyses of spent fuel pool storage risks to support sound decision making by the USNRC and nuclear industry.

1

Introduction

The U.S. Congress[1] asked the National Academy of Sciences[2] (NAS) to conduct a technical study on lessons learned from the Fukushima Daiichi nuclear accident (Sidebar 1.1) for improving safety and security of commercial nuclear power plants[3] in the United States. Congress directed the U.S. Nuclear Regulatory Commission (USNRC) to contract with NAS for this study and also directed that the study "be conducted in coordination with the Department of Energy and, if possible, the Japanese Government" and "build upon the 2004 NAS study of storage issues and complement the other efforts to learn from Fukushima that have already been launched by the [US]NRC and industry."

The 2004 NAS study was also the result of a congressional request made following the September 11, 2001, terrorist attacks on the U.S. homeland. That study examined the safety and security of spent fuel storage in pools and dry casks. Two reports were issued from that study: a classified report (NRC, 2004) and an abbreviated public version of that report (NRC, 2006[4]).

[1] The request is contained in the Consolidated Appropriations Act of 2012 (P.L. 112-74).
[2] Now the National Academies of Sciences, Engineering, and Medicine.
[3] The terms *nuclear power plant* and *nuclear plant* are used interchangeably in this report.
[4] Available for free download from The National Academies Press: http://www.nap.edu/catalog/11263/safety-and-security-of-commercial-spent-nuclear-fuel-storage-public.

SIDEBAR 1.1
Fukushima Daiichi Nuclear Accident

The March 11, 2011, Great East Japan Earthquake and tsunami initiated a severe nuclear accident at the Fukushima Daiichi Nuclear Power Station, located on the northeast coast of Honshu, Japan's largest island (see Figure S1.1). Three of the six reactors at the plant sustained severe core damage, three reactor buildings sustained severe structural damage during the course of the accident, which lasted for several weeks, and four reactor buildings were heavily contaminated with radioactive materials. Radioactive materials released from the damaged reactor cores were transported offsite by winds, resulting in the contamination of parts of several prefectures and the evacuation of more than 100,000 residents. A large portion of the 20-km-radius exclusion zone around the plant will likely to remain off limits to full-time reoccupation for the foreseeable future. See NRC (2014) and the references therein for additional information about the accident and recovery efforts.

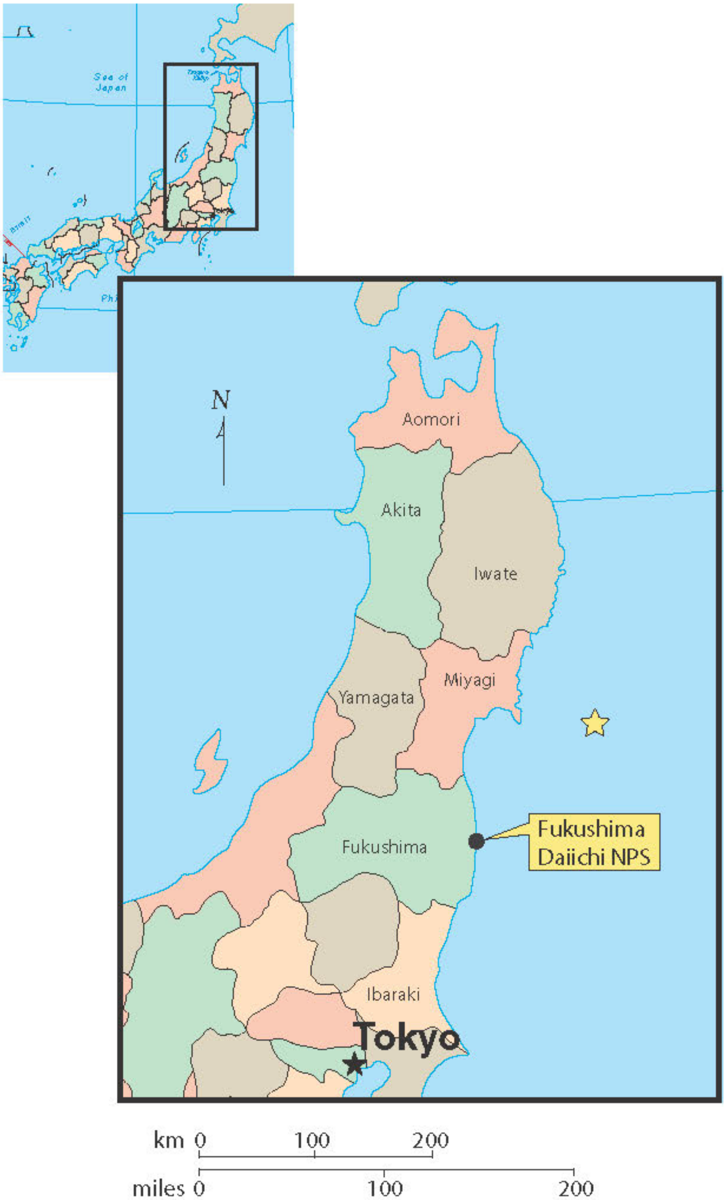

FIGURE S1.1 Map of northern Japan showing the epicenter of the Great East Japan Earthquake (yellow star) and the location of the Fukushima Daiichi Nuclear Power Station.

SIDEBAR 1.2
Statement of Task for This Study

The National Research Council will provide an assessment of lessons learned from the Fukushima nuclear accident for improving the safety and security of nuclear plants in the United States. This assessment will address the following issues:

1. Causes of the Fukushima nuclear accident, particularly with respect to the performance of safety systems and operator response following the earthquake and tsunami;
2. Reevaluation of the conclusions from previous NAS studies on safety and security of spent nuclear fuel and high-level radioactive waste storage, particularly with respect to the safety and security of current storage arrangements and alternative arrangements in which the amount of commercial spent fuel stored in pools is reduced;
3. Lessons that can be learned from the accident to improve commercial nuclear plant safety and security systems and operations;
4. Lessons that can be learned from the accident to improve commercial nuclear plant safety and security regulations, including processes for identifying and applying design-basis events for accidents and terrorist attacks to existing nuclear plants.

The study may examine policy options related to these issues but should not make policy recommendations that involve nontechnical value judgments.

The task for the present study is given in Sidebar 1.2[5] and contains four specific charges:

- Study Charge 1 addresses the causes of the Fukushima Daiichi accident, focusing on the performance of safety systems at the Fukushima Daiichi plant and the responses of its operators following the earthquake and tsunami.
- Study Charge 2 calls for a reevaluation of the conclusions from the 2004 NAS report on spent nuclear fuel safety and security (NRC, 2004). It also calls for an evaluation of current storage arrangements for spent fuel (i.e., pool storage versus dry cask storage) in the context of the 2004 NAS study.
- Study Charges 3 and 4 focus on lessons learned from the Fukushima Daiichi accident for improving safety and security of plant sys-

[5] The statement of task for this study differs in wording from the congressional mandate. NAS shared the revised task with appropriate congressional staff to confirm its acceptability.

tems and operations (Charge 3) and regulations (Charge 4). Study Charge 4 also calls for an assessment of approaches used to identify and apply design-basis events[6] for accidents and terrorist attacks to existing nuclear plants.

An additional sentence was added to the end of the statement of task by NAS to preclude policy recommendations that involve nontechnical value judgments. Such nontechnical factors, for example cost and public acceptability, can be as important as technical factors in the policy-making process.

This study is being carried out in two phases. The phase 1 study report, which was issued in July 2014 (NRC, 2014), addresses the causes of the Fukushima accident and lessons learned for nuclear plant safety. The present report, which provides the results of phase 2 of this study, addresses the following three issues:

(1) Lessons learned from the accident for nuclear plant security.
(2) Lessons learned from the accident for spent fuel storage safety and security.
(3) Reevaluation of conclusions from previous Academies studies on safety and security of spent fuel storage.

1.1 STUDY PROCESS

Phase 2 of this NAS study was carried out by a subset of members from the phase 1 study committee (see NRC, 2014). The phase 2 committee has expertise and experience in several technical disciplines relevant to the study tasks, including geophysics, human factors, law and regulation, materials sciences, mechanical and structural engineering, nuclear engineering, nuclear safety and security, public health, and risk analysis. Biographical sketches of committee and staff members are provided in Appendix A.

The committee held 28 in-person and conference-call meetings during the course of this study to gather information and develop this report. A list of presentations made at the committee's information-gathering meetings is provided in Appendix B. The committee visited several nuclear plants during the first phase of this study, including the Fukushima Daini, Fukushima Daiichi, and Onagawa plants in Japan and the Oyster Creek Generating Station and Edwin I. Hatch Nuclear Plant in the United States (see NRC, 2014, Appendix B). No additional plant visits were made during this

[6] A *design-basis event* is a postulated event that a nuclear plant system, including its structures and components, must be designed and constructed to withstand without a loss of functions necessary to protect public health and safety. Such events are described in NRC (2014). See especially Section 5.2 in Chapter 5.

phase 2 study. The committee also received briefings from the USNRC on aspects of spent fuel safety and security during the first phase of this study.

1.2 STRATEGY FOR ADDRESSING THE STUDY CHARGE

This NAS study is one of many investigations/assessments initiated in the wake of the Fukushima Daiichi accident. Some key written products from these activities are listed in Table 1.1 of NRC (2014). Additional written products have been published since NRC (2014) was released. For example, the International Atomic Energy Agency (IAEA) released its comprehensive assessment of the Fukushima Daiichi accident on August 31, 2015 (IAEA, 2015); the government of Japan submitted an update of Japan's comprehensive report on conditions at the Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Station on September 30, 2015 (Government of Japan, 2015).

Investigations/assessments in the United States were led by the nuclear power industry, through the Institute of Nuclear Power Operations and Nuclear Energy Institute with technical support from plant operators and the Electric Power Research Institute, and also by the U.S. government, primarily through the USNRC with technical support from the Department of Energy and its national laboratories. The work of these groups is described in NRC (2014). The committee used the written products from these activities to inform its work. The peer-reviewed literature also served as an important source of information for this study.

The committee relied almost exclusively on English-language information sources for informing itself on the Fukushima Daiichi accident. English translations of key Japanese government and industry reports were readily available to the committee for this purpose (e.g., see Table 1.1 in NRC, 2014). However, the committee did not have access to the full range of Japanese-language papers, reports, and analyses of the Fukushima Daiichi accident.

There is still a great deal to be learned about the impacts of the accident on the Fukushima Daiichi plant, including impacts on spent fuel storage. Additional information will likely be uncovered as the plant is dismantled and studied, perhaps resulting in new lessons learned and revisions to existing lessons, including those in this report.

As noted in NRC (2014), NAS was asked to carry out a technical assessment of lessons learned from the Fukushima Daiichi accident. NAS was not asked to

- Recommend changes to nuclear plant operations or regulations in Japan or other foreign countries. The mandate from Congress directed NAS to focus on U.S. nuclear plants. However, the committee hopes that the results of this NAS study will be useful to other countries.

- Recommend specific changes to U.S. laws or regulations, for example, to shut down or impose additional operating requirements on reactors in the United States. Such changes are the responsibility of the U.S. government, require the participation of affected stakeholders, and involve consideration of nontechnical factors that are beyond the scope of this study.
- Recommend specific changes to the designs or operations of U.S. nuclear plants. Such changes are the responsibility of the nuclear industry and its regulator, acting in response to their own assessments and with input from interested organizations and individuals, and require plant design-specific information that is unavailable to the committee.
- Assess whether U.S. nuclear plants are safe. The primary focus of this study is on how nuclear plant safety and security can be improved based on lessons learned from the Fukushima Daiichi accident. This focus should not be construed to suggest that nuclear plants are currently unsafe. Nuclear plant operators and regulators strive to make continuous improvements to nuclear plant safety.

1.3 REPORT ORGANIZATION

This report is organized into seven chapters:

- Chapter 1 (this chapter) provides background information about this study.
- Chapter 2 describes the impacts of the March 11, 2011, Great East Japan Earthquake and tsunami on spent fuel storage at the Fukushima Daiichi plant and lessons learned for the United States.
- Chapter 3 identifies lessons that can be learned from the accident to improve commercial nuclear plant security systems, operations, and regulations.
- Chapter 4 provides a reevaluation of the findings and recommendations from previous NAS reports on spent fuel safety and security.

Chapters 5-7 provide supporting information for the reevaluation in Chapter 4:

- Chapter 5 focuses on security risk assessment.
- Chapter 6 focuses of loss-of-cooling events in spent fuel pools.
- Chapter 7 focuses on expedited transfer of spent fuel from pools to dry casks.

Findings and recommendations are provided in Chapters 2, 3, and 4. They are numbered using the format x.z, where x is the chapter number and z is a serial number (1, 2,...).

The appendixes provide committee and staff biographical sketches (Appendix A), a list of presentations made at the committee's information-gathering meetings (Appendix B), conversions and units (Appendix C), and acronyms (Appendix D).

2

Fukushima Daiichi Nuclear Accident: Lessons Learned for Spent Fuel Storage

The focus of this chapter is on the impacts of the March 11, 2011, Great East Japan Earthquake and tsunami on spent fuel storage at the Fukushima Daiichi plant and lessons learned for the United States. This chapter is intended to address Study Charges 3 and 4 (Sidebar 1.2 in Chapter 1) on lessons learned for improving the safety and security of spent fuel storage.

Readers who are not familiar with the Fukushima Daiichi accident may wish to review Chapters 3 and 4 of the committee's phase 1 report (NRC, 2014). See especially Sidebar 3.1 (pp. 90-91) and Table 4.1 (pp. 103-104) in NRC (2014) for a chronology of key accident events. The information presented in this chapter is based primarily on the accident reconstructions by Tokyo Electric Power Company (TEPCO, 2012a) and the Fukushima Daiichi Accident Investigation Committee (Investigation Committee, 2012). Other key sources of information are referenced where used in the chapter.

2.1 SPENT FUEL STORAGE AT THE FUKUSHIMA DAIICHI PLANT

Spent fuel was being stored in eight locations at the Fukushima Daiichi plant on March 11, 2011 (Figure 2.1):

- In pools in each of the six reactor units (Units 1-6),
- In a common pool, and
- In a dry cask storage facility.

Table 2.1 provides information about the quantities of fuel being stored at the plant and the decay heat in the spent fuel pools. The focus of this

FIGURE 2.1 Oblique aerial photo of the Fukushima Daiichi Nuclear Power Station prior to the March 11, 2011, earthquake and tsunami. Note: Spent fuel was being stored in the Unit 1-6 reactor buildings, the common pool, and in a dry cask storage facility. SOURCE: Photo courtesy of TEPCO.

chapter is on the Unit 1-4 spent fuel pools because these units sustained severe damage as a result of the earthquake and tsunami (see NRC, 2014, Chapter 4). Summary information about spent fuel storage in Units 5 and 6, the common pool, and dry cask storage is provided in Appendix 2A.

Units 1-4 at the Fukushima Daiichi plant are boiling water reactors (BWRs) with Mark I containments. Their spent fuel pools are located on the fifth-floor refueling decks in the upper portions of the reactor buildings (Figure 2.2). Each pool (right-hand side of Figure 2.2) is constructed of about 1.5-m (5-ft) thick reinforced concrete with a 5-mm (0.2-inch) thick steel liner. The pools are rectangular in horizontal dimension[1] and almost 12 m deep. Fuel assemblies are stored vertically in metal racks at the bottoms of the pools (Figure 2.3). The racks are covered by about 7 m (23 ft) of water when pool water is at nominal levels. The tops of the spent fuel racks are designed to be slightly taller than the top of the active fuel. The racks in the spent fuel pools at the Fukushima Daiichi plant are about 40 cm taller than the top of the active fuel. In U.S. designs the racks are slightly taller than the top of the active fuel but the exact difference can vary by plant design. The pools contain regions without racking that are used for loading fuel transfer casks.

[1] The Unit 1 pool is 12 m × 7.2 m in area; the Unit 2-4 pools are 12.2 m × 9.9 m in area.

TABLE 2.1 Spent Fuel Storage at the Fukushima Daiichi Plant on March 11, 2011

Unit	Decay Heat (MW)	Water Volume[a] (m^3)	Total Number of Fuel Assemblies in Pool[b]	Fuel Rack Occupancy[c] (%)	Total Radionuclide Activity of Fuel (MCi)	Cs-137 Radionuclide Activity of Fuel[d] (MCi)
1	0.18	990	292 (100)	44	43.2	4.43
2	0.62	1,390	587 (28)	50	149	12.1
3	0.54	1,390	514[e] (52)	46	130	10.6
4	2.26	1,390	1,331 (204)	97	568	23.9
5	1.01	1,390	946 (48)	63	249	—
6	0.87	1,460	876 (64)	53	73.0	—
Common Pool	1.13	3,828	6,375	93	378	—
Dry Cask		N/A	408	N/A		

NOTES: MW = megawatts (10^6 watts); MCi = megacuries (10^6 curies).
[a] Actual pool volumes are slightly larger than the volumes reported here because of the displacement of the fuel and racks and the control of water levels below the top of the pool.
[b] Amounts in parentheses are new fuel assemblies that are in addition to the spent fuel amounts shown.
[c] Including new (unirradiated) fuel.
[d] For comparison, the reactor cores in Units 1, 2, and 3 had Cs-137 activities of 6.5, 6.9, and 6.5 MCi, respectively.
[e] There were mixed oxide fuel assemblies in the Unit 3 reactor but none in the pool.
SOURCES: Activity data from Table 2.1.1-5 of NAIIC (2012) and from estimates by Nishihara et al. (2012) as reported by Povinec et al. (2013); decay heat, water capacity, and assembly data from Attachment 9-1 of TEPCO (2012a).

The pools are located adjacent to the reactor well, a large cavity located above the reactor pressure vessel (Figure 2.2). The well and pool are connected by a short canal that contains two gates.[2] When the gates are closed, the pool and reactor well are hydraulically separated.

The reactor well is located adjacent to the dryer-separator pit (Figure 2.2). That pit is used to store reactor components (i.e., the steam separator, steam dryer, and reactor shroud) when the reactor is undergoing refueling or heavy maintenance. These radioactive components are removed

[2] These gates are approximately 0.9 m (3 ft) wide and 6.4 m (20 ft) tall and have polymeric seals to prevent water leakage. The gate adjacent to the pool is sealed by the pool's water pressure—about 17 tonnes of force when pool water is at nominal levels. See Appendix 2C, especially Figure 2C.1, for details.

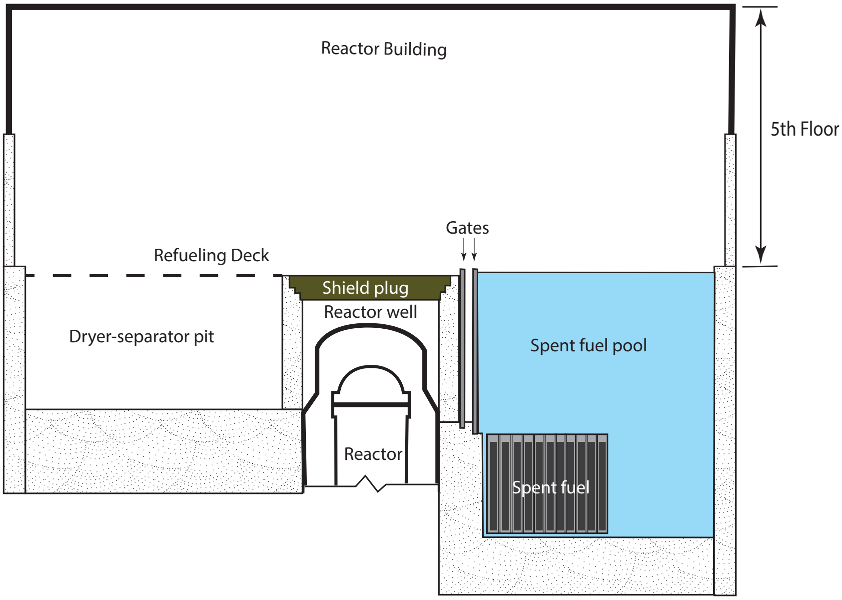

FIGURE 2.2 Schematic cross section of the upper (fifth floor) portion of a BWR Mark I reactor building showing the configuration of the refueling deck in Units 1-3 of the Fukushima Daiichi plant on March 11, 2011. Notes: The features shown in the figure are described in the text. This is not an engineering drawing; some features are simplified or omitted, and not all components are drawn to scale. SOURCE: Adapted from USNRC (2014a, Figure 42).

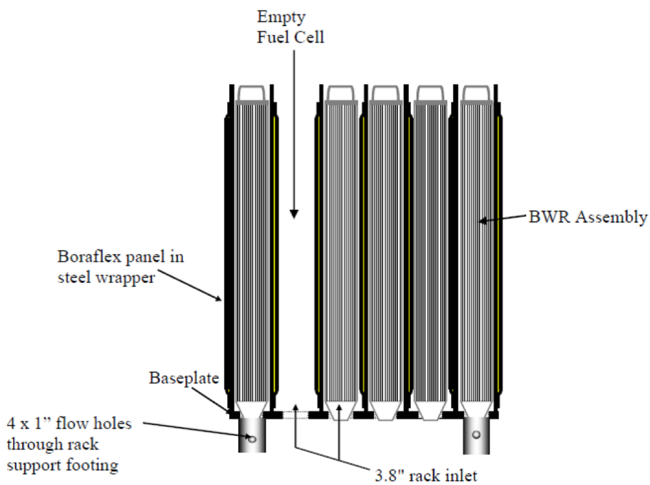

FIGURE 2.3 Typical BWR spent fuel rack with fuel assemblies. Note: The rack is approximately 4.5 m high. SOURCE: Gauntt et al. (2012, Figure 109).

from the reactor and transferred into the pit using an overhead crane that moves along steel rails on the walls of the reactor building.

The reactor well and dryer-separator pit are flooded with water, and the gates are opened for reactor refueling and some maintenance operations (Figure 2.4). Fuel is moved underwater between the reactor and pool using a fuel handling machine that runs on rails located on the refueling floor

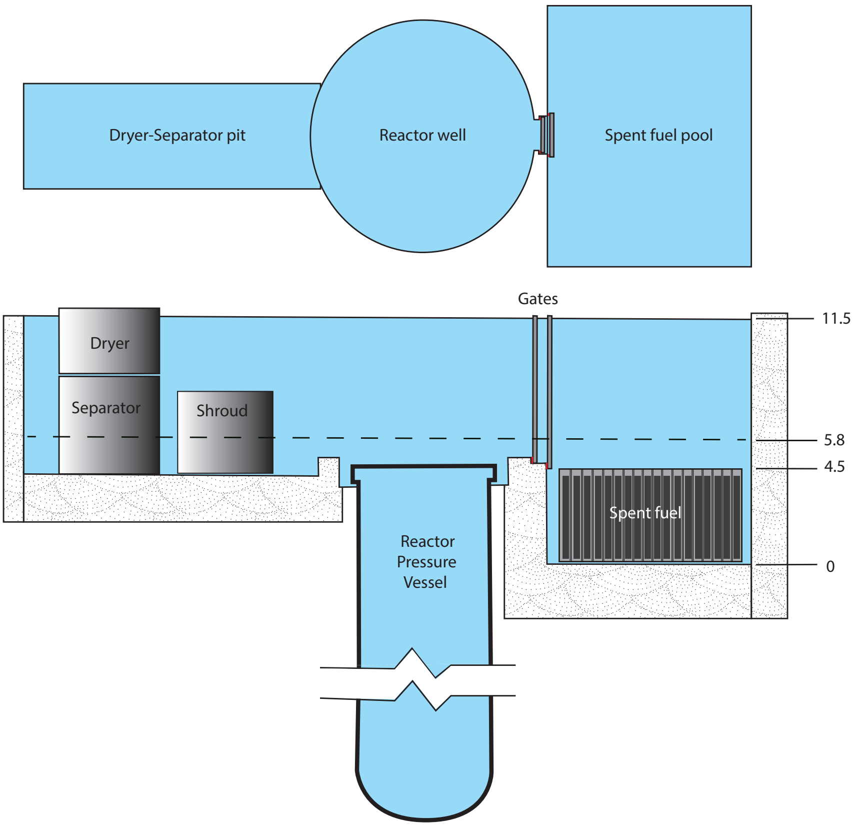

FIGURE 2.4 Schematic cross-sectional (*bottom*) and plan view (*top*) of the Unit 4 refueling deck on March 11, 2011. Notes: The fuel from the reactor was stored in the spent fuel pool, the dryer-separator pit and reactor well were flooded with water, the concrete barriers between the dryer-separator pit and reactor well and the spent fuel pool and reactor well (see Figure 2.2) were removed, and the gates separating the reactor well from the spent fuel pool were closed. Water levels (m) relative to the bottom of the spent fuel pool are shown on the right-hand side of the figure. Note: This is not an engineering drawing; some features are simplified or omitted, and not all components are drawn to scale. SOURCE: Based on information in TEPCO (2012a, Attachment 9) and Investigation Committee (2012, Attachment IV-40).

(Figure 2.5). Spent fuel may be moved from the pool racks into casks for transfer to the common pool prior to reactor refueling to make room in the pool for newly offloaded fuel. These casks are moved in and out of the pool using the overhead crane.

The Unit 1-3 reactors at the Fukushima Daiichi plant were in normal operation prior to the March 11, 2011, earthquake and tsunami. Their refueling decks were configured as shown in Figure 2.2. The gates separating the spent fuel pools from the reactor wells were closed, and their reactor wells and dryer-separator pits were dry.

The Unit 4 reactor was undergoing heavy maintenance on March 11, 2011. Its refueling deck was configured as shown in Figure 2.4. The shield plug and barriers shown in Figure 2.2 between the reactor well and the dryer-separator pit were removed once the reactor well was flooded, the containment vessel and reactor pressure vessel upper heads were removed to access the reactor core, and the gates between the spent fuel pool and reactor well were closed. All of the fuel in the reactor core had been offloaded to the pool. The fuel from the core had high decay heat because the reactor had been shut down only 102 days earlier (Government of Japan, 2011).

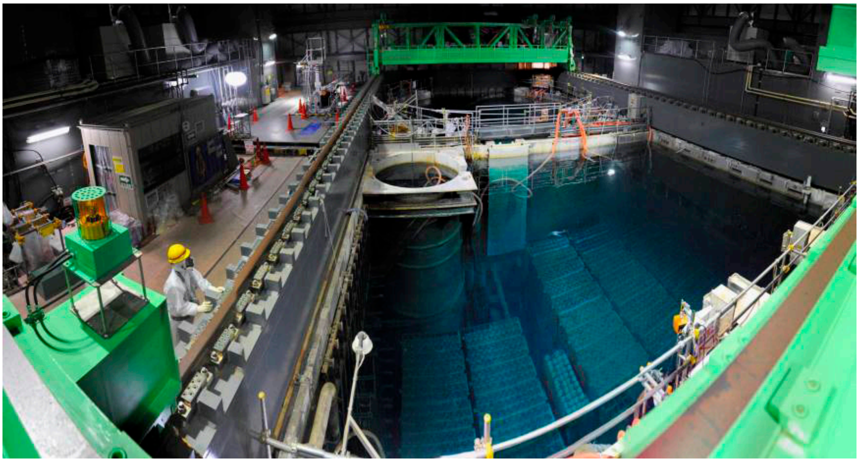

FIGURE 2.5 Photo of the Unit 4 refueling deck at the Fukushima Daiichi plant taken on November 7, 2013. Notes: The spent fuel pool is shown in the foreground; the fuel racks are visible at the bottom of the pool. The reactor well (background) is separated from the pool by the steel gate shown on the far wall of the pool. The green apparatus located above the reactor well is used to lift the gates and runs along tracks on the sides of the pool, The green apparatus that is partially visible in the foreground is the fuel handling machine. It also runs along tracks on the sides of the pool. The photo is distorted because it was taken with an ultra-wide-angle lens. SOURCE: Kyodo News.

The spent fuel pools at Fukushima Daiichi Units 1-4 contained many fewer assemblies than are typically stored in spent fuel pools at U.S. nuclear plants. The storage capacity of U.S. spent fuel pools ranges from fewer than 2,000 assemblies to nearly 5,000 assemblies, with an average storage capacity of approximately 3,000 spent fuel assemblies. U.S. spent fuel pools are typically filled with spent fuel assemblies up to approximately three-quarters of their capacity (USNRC NTTF, 2011, p. 43).

The Unit 1-4 spent fuel pools are equipped with active cooling systems; in particular the Spent Fuel Pool Cooling and Cleanup (FPC) systems, which are located within the reactor buildings below the refueling decks and in a nearby radwaste building. This system is designed to maintain pool temperatures in the range 25°C to 35°C (77°F to 95°F) by pumping the pool water through heat exchangers. The system also filters the pool water and adds makeup water as necessary to maintain pool water levels. All of these features require electrical power.

The pools and refueling levels contain instruments to monitor water levels, temperatures, and air radiation levels. These measurements are displayed in the main control rooms. The temperature and water-level indicators are limited to a few locations[3] near the tops of the pools for the purpose of maintaining appropriate water levels during normal operations:

- Pool water level is monitored by two level switches installed 30 mm (~0.1 ft) above and 160 mm (~0.5 ft) below the normal water level in the pool.
- Pool water temperature is monitored by a sensor 300 mm (~1 ft) below the normal water level of the pool.

This instrumentation also requires electrical power to operate and has no backup power supply.

2.2 IMPACTS OF EARTHQUAKE AND TSUNAMI ON THE UNIT 1-4 SPENT FUEL POOLS

NRC (2014) provides a discussion of key events at the Fukushima Daiichi plant following the March 11, 2011, earthquake and tsunami. To

[3] The description of the instrumentation is what was available in normal operation before March 11, 2011, and is what is nominally provided in typical U.S. BWR Mark I plants. After the tsunami and explosions in Units 1, 3, and 4, the standard equipment was inoperable. In some cases, improvised instrumentation was lowered into the pools, and in other cases the water level was observed indirectly through the overflow to the surge tank and temperature was monitored at selected locations in the FPC systems. The temperature sensor in the pool of Unit 2 was operational once instrument power was restored.

summarize, Units 1-4 lost external power as a result of earthquake-related shaking. Units 1-4 also lost all internal AC power and almost all DC power for reactor cooling functions as a result of tsunami-related flooding. Efforts by plant operators to restore cooling and vent containments in time to avert core damage were unsuccessful. As a result, the Unit 1, 2, and 3 reactors sustained severe core damage and the Unit 1, 3, and 4 reactor buildings were damaged by explosions of combustible gas, primarily hydrogen generated by steam oxidation of zirconium and steel in the reactor core and, secondarily, by hydrogen and carbon monoxide generated by the interaction of the molten core with concrete.

The loss of AC and DC power and cooling functions also affected the Unit 1-4 spent fuel pools: The pools' FPC systems, secondary cooling systems, and pool water-level and temperature instrumentation became inoperable. High radiation levels and explosion hazards prevented plant personnel from accessing the Unit 1-4 refueling decks. Consequently, no data on pool water levels or temperatures were available for almost 2 weeks after the earthquake and tsunami. Moreover, even after pool instrumentation was restored, it was of limited value because of the large swings in pool water levels that occurred during the accident.[4]

Improvised instrumentation and aerial observations were used to monitor pool conditions. Aerial and satellite photography were particularly important sources of information in the early stages of the accident although the images were not always interpreted correctly.

The earthquake caused the reactor buildings to sway, which likely caused water to slosh from the pools.[5] No observational data on sloshing-related water losses are available, however. Analyses performed by the plant owner, TEPCO, suggest that sloshing reduced pool water levels by about 0.5 m (TEPCO, 2012a, Attachment 9-1). The sloshed water spilled onto the refueling decks and likely flowed into the reactor buildings through deck openings such as floor drains.

The explosions in the Unit 1, 3, and 4 reactor buildings likely caused additional water to be sloshed from the pools in those units. Again, no observational data on explosion-related water losses are available. Sloshing due to building motion resulting from the explosions is unlikely to be significant. But sloshing will occur if there is a spatially nonuniform pressure distribution created on the pool surface by an explosion in the region above the pool. This is particularly likely for high-speed explosions that

[4] As noted previously, all of the water-level and temperature instrumentation was installed within 300 mm (~1 ft) of the top of the pools.

[5] Sloshing was directly observed in the Kashiwazaki-Kariwa nuclear plant during a 2007 earthquake (NAIIC, 2012, p. 97) and reportedly resulted in a large loss of water from the spent fuel pools.

create shock or detonation waves.[6] TEPCO estimates that an additional 1 m of water was sloshed from each of the pools as a result of the explosions (TEPCO, 2012a, Attachment 9-1, p. 3/9).

Personnel in the plant's Emergency Response Center (see NRC, 2014, Appendix D) were focused on cooling the Unit 1-3 reactors and managing their containment pressures during the first 48 hours of the accident. They knew that restoring cooling in the spent fuel pools was less urgent and prioritized accordingly. Beginning on March 13, 2011, operators became increasingly concerned about water levels in the pools; their concerns increased following the explosions in the Unit 3 and 4 reactor buildings on March 14 and 15, respectively (Sidebar 2.1).

By the morning of March 15, 2011, it was apparent that the Unit 1-3 reactors had been damaged and were releasing radioactive material. TEPCO evacuated all but about 70 personnel from the plant because of safety concerns (personnel began returning a few hours later). That same day, TEPCO initiated a comprehensive review of efforts to cool the spent fuel pools and made it a priority to determine the status of the Unit 4 pool. TEPCO added the Unit 3 pool to its priority list on the morning of March 16 after steam[7] was observed billowing from the top of the Unit 3 reactor building.

Details about key events in the Unit 1-4 spent fuel pools and operator responses are described in the following sections.

2.2.1 Unit 1 Pool

The explosion in the Unit 1 reactor building on March 12, 2011, blew out the wall panels on the fifth floor, but the steel girders that supported the panels remained intact. The roof collapsed onto the refueling deck and became draped around the crane and refueling machinery[8] (Figure 2.6). This wreckage prevented visual observations of and direct access to the pool.

TEPCO estimated that the pool lost about 129 tonnes of water from the earthquake- and explosion-related sloshing. This lowered the water level in the pool to about 5.5 m above the top of the racks. Because of the very low decay heat in Unit 1[9] (Table 2.1), this pool was of least concern

[6] For example, sloshing has been observed in laboratory experiments with shock wave loading (Teodorcyzk and Shepherd, 2012).

[7] Some descriptions of the accident by TEPCO and others refer to steam emissions from the reactor buildings as "white smoke" emissions.

[8] The refueling floor has been inspected visually (Figure 2.6). Although there is a great deal of rubble, the refueling machine and overhead crane appear to be intact. Rubble removal and demolition will be needed in preparation for installing fuel and cask handling equipment to transfer fuel out of the pool.

[9] All of the spent fuel in the pool was at least 1 year old.

SIDEBAR 2.1
"There is no water in the spent fuel pool"

The explosion that damaged the Unit 4 reactor building on the morning of March 15, 2011, caused extreme anxiety in the media and among Japanese and U.S. government officials. A *New York Times* article[a] on March 15, 2011, featured a satellite photo showing destruction of the upper floors of Units 1, 3, and 4 with a plume of steam rising from Unit 3. The article reported that the explosion in Unit 4 "was caused by hydrogen gas bubbling up from chemical reactions set off by the fuel rods in the pool, Japanese officials said."

Information through official government channels was equally unsettling. Immediately after the earthquake, the USNRC, Department of Energy, and other U.S. agencies sent staff to Tokyo to liaise with and offer assistance to the Japanese government and TEPCO. Agency staff reported back to their agencies by telephone and email, passing along fragmentary and often contradictory information that they were able to obtain from harried Japanese officials who were trying to cope with the earthquake, tsunami, and nuclear disasters.

Early in the morning on March 16, 2011, USNRC staff in Tokyo and the agency's Rockville, Maryland, headquarters held a conference call[b] to discuss conditions at the Fukushima plant. A participant on the call commented that "Right now, Unit 4 doesn't have a spent-fuel pool anymore. It appears that the walls have crumbled and you've just got fuel there." . . . "And Units 1, 2, and 3, it appears that they're at various [water] levels in the spent fuel pool, and it doesn't appear that they're making up the levels."

Other participants on the call were more cautious about making definitive statements about conditions at the plant: "What we're trying to do is get an assessment from the site as to what is the condition of those facilities." And ". . . what we have is a lot of unknowns that we're trying to make known."

Later on March 16, USNRC Chairman Gregory Jaczko provided an update on conditions at the Fukushima plant at a congressional hearing:[c]

> "What we believe at this time is that there has been a hydrogen explosion in this unit [Unit 4] due to an uncovering of the fuel in the fuel pool. We believe that secondary containment has been destroyed and there is no water in the spent fuel pool. And we believe that radiation levels

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[a] *New York Times*. Japan Says 2nd Reactor May Have Ruptured With Radioactive Release. March 15, 2011. Available at http://www.nytimes.com/2011/03/16/world/asia/16nuclear.html.

[b] Transcript of USNRC telephone communications on March 16, 2011. USNRC Freedom of Information Act (FOIA) response, Document No. ML12052A108. Available at http://pbadupws.nrc.gov/docs/ML1205/ML12052A108.pdf.

[c] Transcript of the Joint Hearing Before The Subcommittee on Energy and Power and the Subcommittee on Environment and the Economy of the Committee on Energy and Commerce House of Representatives, One Hundred Twelfth Congress, March 16, 2011. Serial No. 112-20. Available at https://www.gpo.gov/fdsys/pkg/CHRG-112hhrg68480/pdf/CHRG-112hhrg68480.pdf.

> are extremely high, which could possibly impact the ability to take corrective measures."

Chairman Jaczko also described the advice that had been given to the U.S. Ambassador to Japan earlier that day:

> "Recently, the [US]NRC made a recommendation that based on the available information that we have, that for a comparable situation in the United States, we would recommend an evacuation to a much larger radius than has currently been provided in Japan. As a result of this recommendation, the ambassador in Japan has issued a statement to American citizens that we believe it is appropriate to evacuate to a larger distance, up to approximately 50 miles."[d]

At that time, the Japanese government had issued a 20-km (~12-mile) evacuation directive to its citizens.

Chairman Jaczko's testimony sparked a strong reaction in the media:[e] ". . . this, folks, is the nightmare scenario we've been telling you about for days now." The media also recognized the political implications of the conflicting evacuation recommendations issued by the U.S. and Japanese governments: "It's astounding. It is shocking. You know this is a—this is an earthquake of a different kind. This is a political earthquake."

The USNRC was not the only organization to worry about insufficient water in the Unit 4 pool. Japan's Nuclear and Industrial Safety Agency (NISA) also reached this conclusion as noted in the chapter text.

The Japanese government issued a measured public response to the U.S. government announcements about conditions in the Unit 4 spent fuel pool:[f]

> "Because we have been unable to go the scene, we cannot confirm whether there is water left or not in the spent fuel pool at Reactor No. 4." (Yoshitaka Nagayama, NISA)

As did TEPCO:

> "We can't get inside to check, but we've been carefully watching the building's environs, and there has not been any particular problem." (Hajime Motojuku, TEPCO)

continued

[d] Ultimately, evacuations were ordered by the Japanese government in certain locations as far as 50 km (31 miles) from the plant. The basis for the NRC recommendation is discussed in the June 17, 2011, letter from Chairman Jaczko to Senator Webb; see http://www.nrc.gov/reading-rm/doc-collections/congress-docs/correspondence/2011/webb-06-17-2011.pdf.

[e] Quotes are taken from a CNN broadcast on March 16, 2011. Transcript available at http://www.cnn.com/TRANSCRIPTS/1103/16/ita.01.html.

[f] *New York Times*. U.S. Calls Radiation 'Extremely High;' Sees Japan Nuclear Crisis Worsening. March 16, 2011. Available at http://www.nytimes.com/2011/03/17/world/asia/17nuclear.html.

SIDEBAR 2.1 Continued

TEPCO was sufficiently concerned about the conditions of the pools that it obtained video footage and still images (Figure S2.1) during a helicopter overflight of the plant on the afternoon[g] of March 16, 2011. The written accounts (Investigation Committee, 2011; RJIF, 2014; TEPCO, 2012a) all report that TEPCO was sufficiently convinced by the visual evidence of sunlight reflecting from the water surface that it announced at a press conference at noon on March 17 (Japan Standard Time) that it had confirmed that there was water covering the spent fuel in the Unit 4 pool. However, over 3 weeks elapsed before instruments were lowered from a boom of the concrete pump truck to obtain the first reliable measurement of water levels.

This episode illustrates how difficult it is to make evacuation decisions in the absence of accurate and timely information about the condition of the plant and the progress of ongoing mitigation efforts. It also illustrates the communication challenges that can arise during a crisis: TEPCO was managing a multireactor emergency and had little information about the condition of the Unit 1-4 spent fuel pools during the first few days of the accident. The destruction of physical infrastructure by the earthquake and tsunami and heavy demands on TEPCO and Japanese officials made it difficult for U.S. personnel in Tokyo to obtain timely and reliable information.

FIGURE S2.1 Photograph of Unit 4 obtained during the March 16, 2011, helicopter overflight of the Fukushima Daiichi site. The photo shows the refueling deck region in the vicinity of the spent fuel pool. The green equipment visible in the center of the image is the fuel handling machine. SOURCE: http://photo.tepco.co.jp/en/date/2011/201103-e/110317-01e.html.

[g] Japan Standard Time. This flight took place early in the morning on March 16, 2011, U.S. Eastern Standard Time. Videos and photos taken from the overflight are available at http://photo.tepco.co.jp/en/date/2011/201103-e/110317-01e.html.

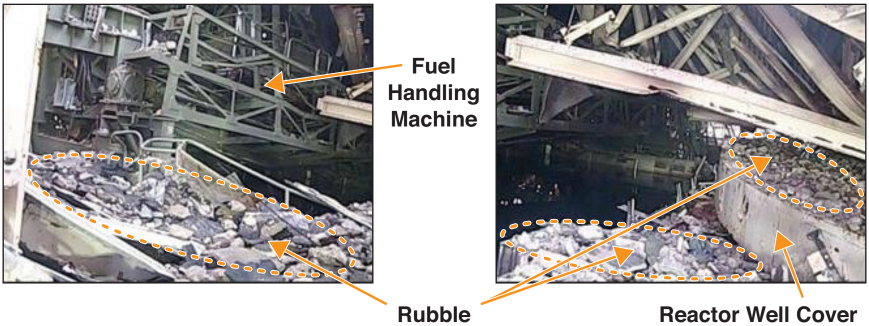

FIGURE 2.6 Views of the damage on the refueling deck in Unit 1. Note: See http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d141225_10-j.pdf for more images. SOURCE: Tateiwa (2015).

to plant operators, and in fact water was not added to the pool until the end of March 2011.

Operators initially considered several approaches for adding water to the Unit 1 pool: sending personnel into the reactor building to connect fire hoses to an FPC line, dropping ice or water from helicopters into the pool, and spraying water into the pool using a fire truck equipped with a ladder. All of these approaches were deemed to be impractical or pose unacceptable risks to plant personnel.

Operators used a concrete pump truck to add fresh water to the pool starting on March 31, 2011. Water was pumped onto the collapsed roof structure and some of it ran off into the pool. The amount of water that reached the pool was uncertain because the pool could not be observed visually.

A subsequent survey of the Unit 1 reactor building indicated that radiation levels were relatively low at the southwest corner of the third floor where the FPC pump and heat exchanger were located. Operators gained access to that area and were able to remove a check valve in the FPC piping and attach a temporary adapter for a fire hose. The fire hose was run outside the building to a temporary motor-driven pump connected to fresh-water storage tanks. Water was injected into the pool by this method after May 28, 2011.

An alternate cooling system was installed and put into operation on August 10, 2011. This system used the spent fuel pool heat exchanger and pump and a temporary air-cooled heat exchanger and closed-loop circulation system outside the building.

TEPCO estimated that the water level in the Unit 1 pool decreased by 3 m (to about 4 m above the top of the racks) by May 21, 2011: 1.5 m of this water loss was due to sloshing and another 2.2 m of water loss was due to evaporation; credit is taken for 0.7 m (63 tonnes) of water added on March 31.

The committee estimated the evaporative water loss in the Unit 1 pool using the steady-state energy-balance model described in Appendix 2B. The main contribution of the committee's model was to examine in detail the role of leakage through the gates between the reactor well and the spent fuel pool in Unit 4, a key issue that was identified during the course of the committee's deliberations. The evaporative loss from Units 1, 2, and 3 was also examined but in much less detail.

The committee estimates that evaporative water losses from the Unit 1 pool between March 11 and May 21, 2011, totaled about 2.6 m, resulting in a water level about 4.4 m above the top of the racks. If TEPCO's sloshing loss estimates are included and no credit is given for water additions by the concrete pump truck, then water levels could have been as low as 2.9 m above the top of the racks.[10]

2.2.2 Unit 2 Pool

There was no explosion in the Unit 2 reactor building, but a blowout panel on the east side of the refueling deck became dislodged after the Unit 1 explosion on March 12, 2011. Steam was occasionally observed to emerge from this opening. Its origin is unclear, but it may have originated from the interior of the reactor building rather than from the spent fuel pool. There has been no evaluation of how this opening affected the evaporation rate from the pool.[11]

Operators focused their efforts on using components of the installed FPC system to inject water into the pool. The initial idea was to inject water through the Make-Up Water Condensate system line. However, this would require the replacement of a power panel and pump inside the contaminated turbine building where radiation levels were deemed too high to work.

Instead, plant personnel removed a sight glass from the FPC line in the radioactive waste building and connected a fire hose. Seawater from the north quay was pumped through this hose using a fire truck (and later a motor-driven pump).

Seawater was injected into the pool starting on March 20, 2011, and continued intermittently thereafter. The water supply was switched to

[10] This water-level estimate is consistent with TEPCO's estimate with no credit given for water injection on March 31, 2011 (TEPCO, 2012a, Attachment 9-2).

[11] Simulations using Methods for Estimation of Leakages and Consequences of Releases (MELCOR; Gauntt et al., 2012) predict that the net evaporation rate is reduced by a factor of 2 when a pool is enclosed in an intact building relative to when there is no enclosure. (Evaporated water condenses on the building and structures with some fraction flowing back into the pool.) Gauntt et al. did not simulate the effect of partial building venting.

fresh water on March 29 and pumping at regular intervals continued until May 31. At that time cooling was switched over to an alternate cooling system that used the FPC lines but with a new primary heat exchanger and pump in the radioactive waste building and a new secondary cooling system with a cooling tower outside the building. Between 40 and 60 tonnes of water were injected into the pool every 3 days until the end of May 2011.

TEPCO estimates that water levels in the pool had dropped to about 5.5 m above the top of the racks by the end of March 2011. However, there are no measurements of actual pool levels. According to the committee's steady-state energy-balance model (Appendix 2B), pool levels would have decreased by about 0.16 m/day due to evaporation, which is consistent with the observed temperature records[12] from TEPCO (TEPCO, 2012a, Attachment 9-3, pp. 7/8-8/8).

Measured water temperatures in the pool fluctuated between 50°C and 70°C with the periodic addition of water to maintain pool levels. The temperature fluctuations were likely caused by changes in pool water levels: When the pool was full, the temperature sensor was submerged and indicated the water temperature; when the water level dropped, the sensor was exposed and indicated the air-vapor temperature near the top of the pool.

2.2.3 Unit 3 Pool

The explosion in the Unit 3 reactor building on March 14, 2011, damaged the northwest side of the fifth floor, collapsing the steel and concrete structure. Steel framing and concrete members were collapsed on top of each other on the fourth floor, and substantial portions of the walls of the fourth floor were also damaged.[13] The explosion introduced a substantial amount of debris, building structural components, and equipment (including the fuel handling machine) into the spent fuel pool. These materials are resting on the bottom of the pool and on the top of the racks (Figures 2.7 and 2.8).[14]

Steam was observed billowing from the top of Unit 3 on the morning of March 16, 2011. This steam, combined with the extensive destruction of the fourth and fifth floors, suggested to some observers that the pool was damaged or low on water and the spent fuel was undergoing rapid steam oxidation (see Sidebar 2.2).

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[12] As noted previously, the temperature gauge is located 300 mm (1 ft) below the top of the normal water level in the pool. Temperature readings indicate that the gauge was uncovered in 1-2 days, indicating that pool water level decreased by about 0.2 m per day.

[13] Radioactive contamination in the building has prevented the onsite assessment of the integrity of the structure under the pool.

[14] Debris has been cleared from the refueling deck, and work is now in progress to clear debris from the pool to assess the condition of the fuel and racks. TEPCO announced on August 2, 2015, that it had removed the spent fuel handling machine from the pool. See http://www.tepco.co.jp/en/press/corp-com/release/2015/1256671_6844.html.

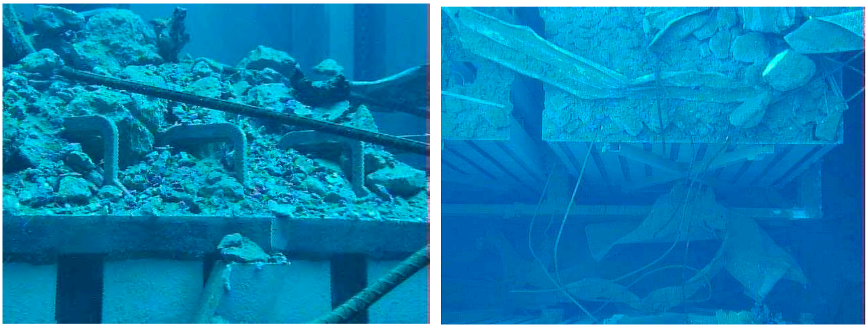

FIGURE 2.7 Images of debris in Unit 3 spent fuel pool on top of and beside the racks. *Left image:* Side view of the tops of the racks. Three fuel assembly handles are partially visible in the foreground of the image. *Right image:* Top view of the racks. SOURCE: TEPCO, used in NRA (2014, Figure 6.1).

A TEPCO-arranged helicopter flight over the Fukushima Daiichi site on the afternoon of March 16, 2011, obtained additional information about conditions in the pools. Debris and billowing steam obscured the Unit 3 pool, so little visual information could be obtained about its condition. A later analysis[15] of aerial photographs and thermal imaging was undertaken by the Nuclear Regulation Authority (NRA, 2014) to pinpoint the origin of the steam: It was found to originate from near the reactor well cover and the dryer-separator pit, not the spent fuel pool. Thermal imaging on March 20, 2011, showed that the pool water temperature was about 60°C, whereas temperatures adjacent to the reactor cover and dryer-separator pit measured by thermal imaging were over 100°C. (Building debris obscured the exact source of the steam plume.) The steam may have been generated in the reactor pressure vessel or containment.

Operators made several efforts to add water to the Unit 3 spent fuel pool following the helicopter overflight. From March 17 to March 25, 2011, operators attempted to add water using helicopters and fire trucks. The helicopter water drops were unsuccessful, and the fire truck sprays were ineffective. On March 23 and 24, an attempt was made to inject sea water through an improvised connection to the FPC system. This also was ineffective, apparently because the lines were clogged. From March 27 to April 22, operators added water using a concrete pump truck. The strainer in an FPC line was subsequently removed, and water was successfully injected into the pool beginning on April 26 and continuing until an

[15] This analysis was undertaken to clear up questions that had been raised in the July 2012 Diet report on the accident (NAIIC, 2012).

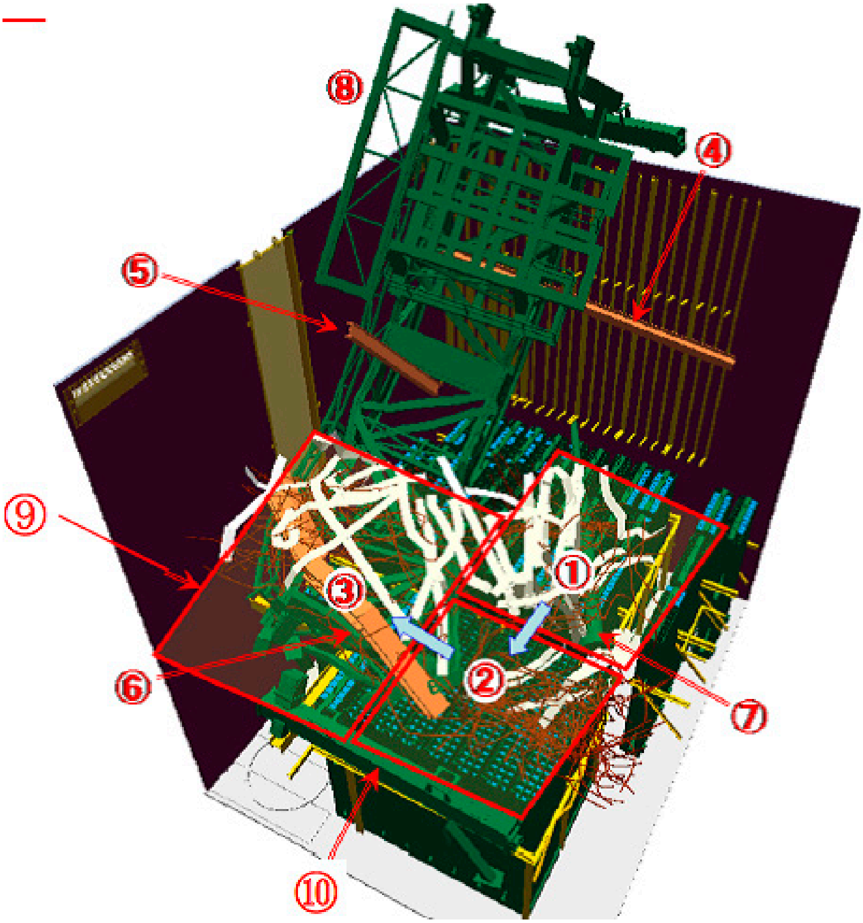

FIGURE 2.8 Location of large structural components in Unit 3 spent fuel pool. Note: The large green object #8 is the fuel handling machine. SOURCE: http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d141225_10-j.pdf.

alternate cooling system similar to that used in Unit 2 (i.e., connecting a fire hose to the FPC system in the radwaste building and pumping seawater using a fire truck) was established on June 30.

The committee's steady-state energy-balance model (Appendix 2B) estimates that pool levels dropped by about 2.5 m between March 11 and April 2, 2011, lowering the water level in the pool to about 4.5 m above the top of the racks. If TEPCO's sloshing losses are included, then water levels could have been as low as 3.0 m above the top of the racks if no

SIDEBAR 2.2
Spent Fuel Heat-up Following Loss-of-Pool-Coolant Events

Spent fuel continues to generate heat from the decay of its radioactive constituents long after it is removed from a reactor. The fuel is stored in water-filled pools (i.e., *spent fuel pools*) to provide cooling and radiation shielding. An accident or terrorist attack that damaged a spent fuel pool could result in a partial or complete loss of water coolant. Such *loss-of-pool-coolant events* can cause the fuel to overheat, resulting in damage to the metal (zirconium) cladding of the fuel rods and the uranium fuel pellets within and the release of radioactive constituents to the environment.

The loss of water coolant from the pool would cause temperatures in the stored spent fuel to increase because air is a less effective coolant than water. The magnitude and rate of temperature increase depends on several factors, including how long the fuel has been out of the reactor and the rate and extent of water loss from the pool.

As fuel temperatures rise, internal pressures in the fuel rods will increase and the rod material will soften. At about 800°C, internal pressures in the fuel rod will exceed its yield stress, resulting in failure, a process known as *fuel ballooning*. Thermal creep of the fuel rod above about 700°C can also result in ballooning.

Once the fuel cladding fails, the gaseous and volatile fission products stored in the gap between the fuel rod and pellets will be released. The fission product inventory varies depending on the type of fuel and its irradiation history; typically, on the order of a few percent of the total noble gas inventory (xenon, krypton), halogens (iodine, bromine), and alkali metals (cesium, rubidium) present in the fuel will be released.

Between about 900°C and 1200°C, highly exothermic chemical reactions between the fuel rods and steam or air will begin to accelerate, producing zirconium oxide:

$$Zr + O_2 \rightarrow ZrO_2 \qquad \text{heat released} = 1.2\times10^7 \text{ joules/kilogram}$$
$$Zr + 2H_2O_{steam} \rightarrow ZrO_2 + 2H_2 \qquad \text{heat released} = 5.8\times10^6 \text{ joules/kilogram}$$

The reaction in steam also generates large quantities of hydrogen. Deflagration (i.e., rapid combustion) of this hydrogen inside the spent fuel pool building can damage the structure and provide a pathway for radioactive material releases into the environment. Further temperature increases can drive more volatile fissile products out of the fuel pellets and cause the fuel rods to buckle, resulting in the physical relocation of rod segments and the dispersal of fuel pellets within the pool.

At about 1200°C the oxidation reaction will become self-sustaining, fully consuming the fuel rod cladding in a short time period if sufficient oxygen is available (e.g., from openings in the spent fuel pool building) and producing uncontrolled (runaway) temperature increases. This rapid and self-sustaining oxidation reaction, sometimes referred to as a *zirconium cladding fire*, may propagate to other fuel assemblies in the pool. In the extreme, such fires can produce enough heat to melt the fuel pellets and release most of their fission product inventories.

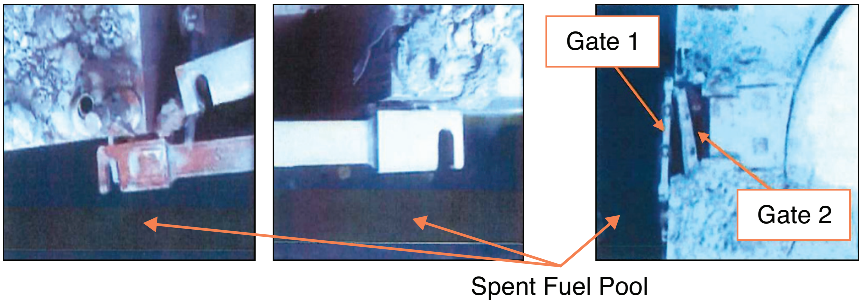

FIGURE 2.9 Overhead photo of the spent fuel gates in the Unit 3 spent fuel pool. Note: The pool is the dark region at the bottom of the two left-hand images and the left-hand side of the right-hand image. The canal and reactor well plug are visible in the center and right-hand sides of the right-hand image. The gate closest to the canal has been displaced. SOURCE: http://www.tepco.co.jp/nu/fukushima-np/handouts/2015/images/handouts_150406_04-j.pdf.

credit is given for the water added by the concrete pump truck beginning on March 27, 2011.

Recent visual inspections of the pool reveal that one of the spent fuel pool gates has been displaced (see Figure 2.9) from its normal position and appears to be deformed. It is not clear whether this displacement was caused by the earthquake, the explosion, or debris falls.

2.2.4 Unit 4 Pool

The Unit 4 reactor was shut down for maintenance, and large-scale repairs were in progress (see NRC, 2014, Chapter 4) on March 11, 2011. The configuration of the Unit 4 refueling deck is illustrated in Figure 2.4: The reactor well and dryer-separator pit had been flooded with water, internal reactor components (steam separator, steam dryer, and reactor shroud) had been removed from the reactor and placed in the pit, and the reactor core comprising 548 fuel assemblies had been removed from the reactor and placed into contiguous racks in the spent fuel pool (Figure 2.10). The gates[16] separating the reactor well from the spent fuel pool were closed.

The explosion that occurred in the Unit 4 reactor building at 06:14 on March 15, 2011, destroyed the roof and most of the walls on the fourth and fifth (refueling deck) floors, and it damaged some of the walls on the third floor. TEPCO (2012a) has suggested that the explosion was due to the com-

[16] The configuration for the two gates shown for Unit 3 (Figure 2.9) appear to be similar to the configuration in Unit 4 according to TEPCO (2012a, Figure 3 of Attachment 9-5); see Figure 2C.1 in Appendix A.

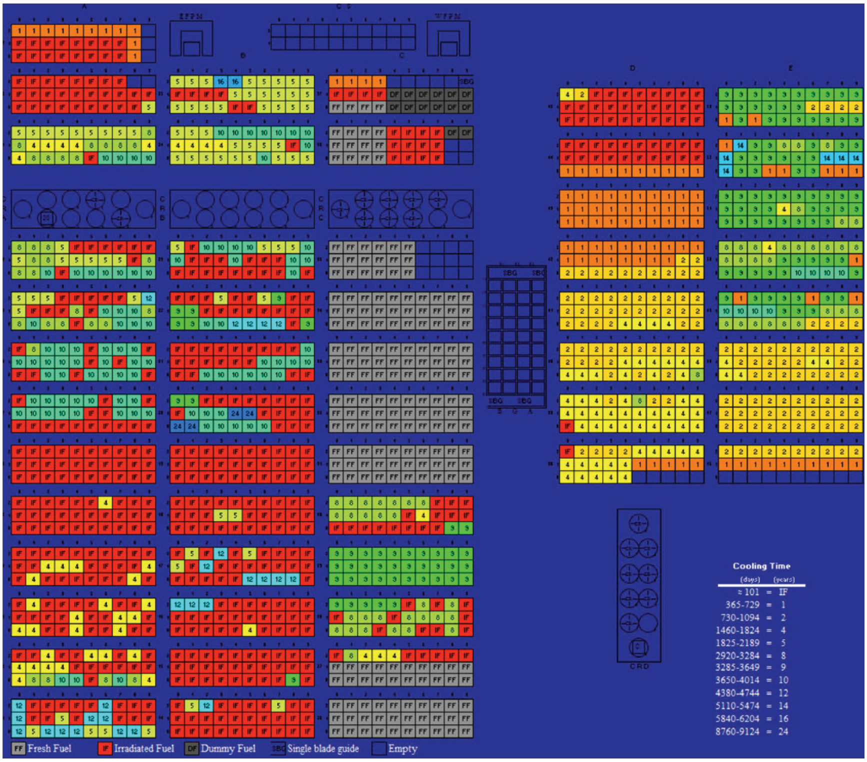

FIGURE 2.10 Fuel thermal loading in Unit 4 pool. Notes: The red squares indicate the hottest, most recently offloaded fuel assemblies. The grey squares indicate unirradiated (new) fuel assemblies. SOURCE: Wang et al. (2012, Figure 2). Copyright 2012 by the American Nuclear Society, La Grange Park, Illinois.

bustion of hydrogen that was generated in Unit 3 and flowed into Unit 4 through the ventilation system.[17] The fifth-floor slab was pushed upward and the fourth-floor slab was depressed.[18] The explosion also deposited

[17] As mentioned previously, carbon monoxide from core-concrete interaction in Unit 3 is another potential source for combustible gas that could have fueled the explosion in Unit 4.

[18] Plant operators were concerned about the integrity of the building structure underneath the Unit 4 pool following the explosion. The building was surveyed, concrete was tested, and the building response to an earthquake was simulated as part of the assessment of the structure. In June 2011, the region underneath the pool was reinforced with steel beams and filled with concrete; this work was completed on July 30, 2011. Quarterly inspections have been carried out since May 2012 and no significant issues have been found. The Unit 4 pool's inventory of 1,535 fuel assemblies was moved into the common pool (spent fuel) or Unit 6 pool (new fuel) between November 2013 and December 2014.

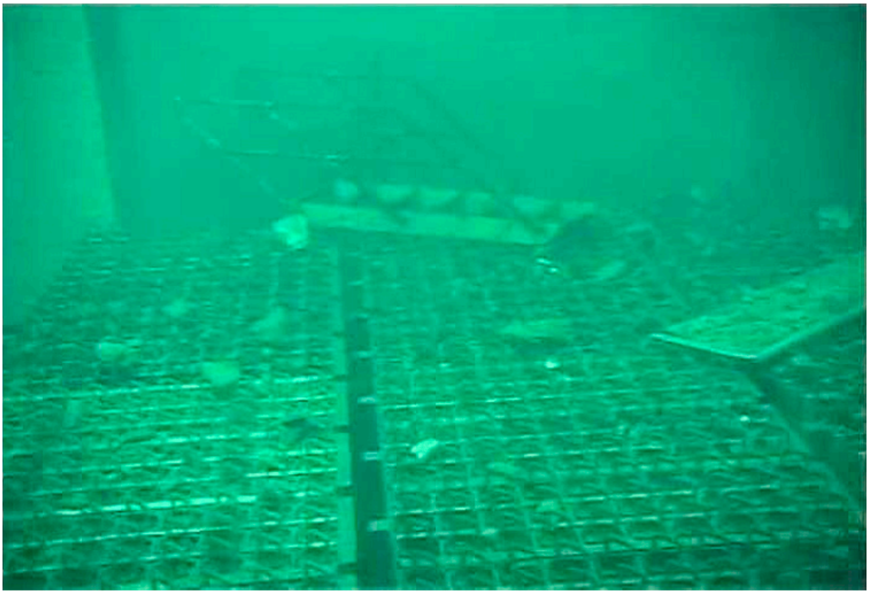

FIGURE 2.11 Debris on top of fuel racks in Unit 4 pool. SOURCE: TEPCO (2012a, Attachment 9-5, Figure 7).

debris around the reactor building, onto the refueling deck, and into the pool (Figure 2.11). Fires were reported in the damaged building later that morning and on the morning of March 16; these fires self-extinguished and were later attributed to the ignition of lubricating oil.

The damage to the Unit 3 and 4 building structures and steam emissions from both buildings raised grave concerns about the spent fuel pools in those units (see Sidebars 2.1 and 2.3). Unit 4 was of particular concern because the reactor contained no fuel and therefore could not have been the source of hydrogen or other combustible gas. The only apparent source of combustible gas within Unit 4 was hydrogen from the steam oxidation of spent fuel in the fully or partially drained Unit 4 spent fuel pool (Sidebar 2.4).

Plant operators well understood the hazard posed by the spent fuel in the Unit 4 pool: The pool was loaded with high-decay-heat fuel; its water level was dropping because of large evaporative water losses; and openings in the Unit 4 building created by the explosion created pathways for radioactive materials releases into the environment. Operators communicated this understanding to TEPCO headquarters and to government agencies. However, there were disagreements between TEPCO and regulators about the status of the Unit 4 pool:

SIDEBAR 2.3
The Devil's Scenario

By late March 2011—some 2 weeks after the earthquake and tsunami struck the Fukushima Daiichi plant—it was far from obvious that the accident was under control and the worst was over. Chief Cabinet Secretary Yukio Edano feared that radioactive material releases from the Fukushima Daiichi plant and its sister plant (Fukushima Daini) located some 12 km south could threaten the entire population of eastern Japan:

> "That was the devil's scenario that was on my mind. Common sense dictated that, if that came to pass, then it was the end of Tokyo." (RJIF, 2014)

Prime Minister Naoto Kan asked Dr. Shunsuke Kondo, then-chairman of the Japanese Atomic Energy Commission, to prepare a report on worst-case scenarios from the accident[a] (see Chapter 3, p. 58, of NAIIC [2012] for a discussion of the circumstances of that report). Dr. Kondo led a 3-day study involving other Japanese experts and submitted his report (Kondo, 2011) to the prime minister on March 25, 2011. The existence of the report was initially kept secret because of the frightening nature of the scenarios it described. An article in the *Japan Times*[b] quoted a senior government official as saying, "The content [of the report] was so shocking that we decided to treat it as if it didn't exist." When the existence of the document was finally acknowledged in January 2012, Special Advisor (to the Prime Minister) Goshi Hosono stated: "Because we were told there would be enough time to evacuate residents (even in a worst-case scenario), we refrained from disclosing the document due to fear it would cause unnecessary anxiety (among the public). . . ."

One of the scenarios involved a self-sustaining zirconium cladding fire in the Unit 4 spent fuel pool. Radioactive material releases from the fire were estimated to cause extensive contamination of a 50- to 70-km region around the Fukushima Daiichi plant with hotspots significant enough to require evacuations up to 110 km from the plant. Voluntary evacuations were envisioned out to 200 km because of elevated dose levels. If release from other spent fuel pools occurred, then contamination could extend as far as Tokyo, requiring compulsory evacuations

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[a] NAIIC (2012) describes the circumstances of the report (see p. 58). The report content is described in a Reuters article that is available at http://www.reuters.com/article/2012/02/17/us-japan-nuclear-scenarios-idUSTRE81G09120120217.

[b] *Japan Times.* Cabinet Kept Alarming Nuke Report Secret. January 22, 2012. Available at http://www.japantimes.co.jp/news/2012/01/22/news/cabinet-kept-alarming-nuke-report-secret/#.VOjYwC4nIQs.

out to more than 170 km and voluntary evacuations out to more than 250 km; the latter includes a portion of the Tokyo area. There was particular concern that the zirconium cladding fire could produce enough heat to melt the stored fuel, allowing it to flow to the bottom of the pool, melt through the pool liner and concrete bottom, and flow into the reactor building. After leaving office, Prime Minister Kan stated that his greatest fears during the crisis were about the Unit 4 spent fuel pool (RJIF, 2014).

A numerical simulation of atmospheric dispersions was carried out by the National Atmospheric Release Advisory Center (NARAC) at Lawrence Livermore National Laboratory on March 20, 2011 (Bader, 2012; Sugiyama et al., 2013[c]). This study assumed a source term of the entire core of the Unit 2 reactor over 24 hours as well as 50 percent of the Unit 3 spent fuel pool and 100 percent of the Unit 4 spent fuel pool over 48 hours. The dispersion was computed using a numerical atmospheric weather model and meteorological data for the region between Tokyo and Fukushima on March 14, 2011, for a 4-hour period on March 14, 2011, and repeated that weather pattern so that the plume primarily headed toward Tokyo over the duration of the simulation. The estimated 4-day total effective dose to individuals at the U.S. Embassy in Tokyo was estimated to be 1.1 rem, mostly from cesium-137. At this dose, the U.S. Environmental Protection Agency recommends that consideration should be given to evacuation or sheltering in place.

This dose rate, extrapolated over a year, would exceed the 2-rem protective action guideline established by the U.S. Environmental Protection Agency for consideration of long-term relocation. However, in considering long-term relocation, other factors such as the potential for decontamination and the uniformity (or spottiness) of the contamination may also play a role in the decision to relocate versus mitigate. The protective action guideline is used to trigger consideration of all the various actions that may be taken to mitigate dose consequences up to and including relocation.

This episode illustrates how uncertainties about the state of the Fukushima Daiichi plant and future progression of the accident—weeks after the accident was initiated—complicated planning and decision making for Japanese officials who were trying to protect the public from potential offsite releases.

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[c] Scenario estimates are available at http://pbadupws.nrc.gov/docs/ML1406/ML14064A101.pdf, pp. 476-479, and in the FOIA document provided to the committee by E. Lyman, Union of Concerned Scientists, May 26, 2015. A description of the circumstances leading to NARAC's involvement is available at http://www.slate.com/articles/health_and_science/science/2013/09/fukushima_disaster_new_information_about_worst_case_scenarios.2.html. For another perspective on the issues around the technical assessments of the disaster and policy decisions see http://japanfocus.org/-Kyle-Cleveland/4116/article.html.

SIDEBAR 2.4
Hydrogen Production in the Unit 4 Spent Fuel Pool: A Hypothetical Scenario

Loss-of-coolant accidents in spent fuel pools can result in the production of sufficient hydrogen to create combustion hazards in spent fuel pool buildings. To illustrate, consider how little oxidation would be required to produce a combustible hydrogen-air mixture in the building covering the refueling deck of Unit 4 at the Fukushima Daiichi plant had the loss of cooling in the pool resulted in the metal-steam oxidation reaction described in Sidebar 2.2.

The oxidation of as little as 3.4 percent of the zirconium inventory in the pool—equivalent to the zirconium contained in about 45 fuel assemblies—by the metal-water reaction would have been sufficient to produce a flammable hydrogen-air mixture in the spent fuel pool building (see Table S2.1). The complete consumption of zirconium in the pool would have produced about 5,000 kg of hydrogen, enough to fill a volume of 62,000 m^3 at standard temperature and pressure. This is almost three times the volume of the Unit 4 building above the refueling deck.

TABLE S2.1 Estimated Mass of Hydrogen Production by Oxidation of Spent Fuel Assemblies in the Unit 4 Spent Fuel Pool at the Fukushima Daiichi Plant

Zr Oxidation[a] (%)	H_2 Produced (kg)	H_2 (vol %)[b]	Zr (tonne)	No. 9 × 9 Fuel Assemblies[c]
1.7	86	5	2.0	23
3.4	171	10[d]	3.9	45
5.1	256	15	5.8	68
6.8	341	20	7.7	90
8.5	426	25	9.7	113
10.2	512	30	11.6	135

[a] Percentage of zirconium inventory in the pool that is oxidized by the metal-water reaction shown in Sidebar 2.2.
[b] Volume fraction of hydrogen contained in the hydrogen-air mixture in the assumed refueling bay volume of 21,000 m^3.
[c] Assuming 86 kg of zirconium per assembly for the 1,331 spent fuel assemblies.
[d] Combustion may occur above 4 percent in a dry mixture; 10 percent is used as the assumed flammability limit in MELCOR simulations.
SOURCE: Computations by committee based on stoichiometry of reaction.

> ". . . NISA [Nuclear and Industrial Safety Agency of Japan] and the [US] NRC both insisted that the water level in the Unit 4 SFP [spent fuel pool] had dropped and the fuel was exposed, but TEPCO insisted that the fuel was not exposed because as of 15:00 [on March 15] when the explosion at the Unit 4 reactor building occurred, not enough heat was being generated to cause the fuel to be exposed, and surrounding radiation levels were too low to indicate that the fuel was exposed." (TEPCO, 2012a, p. 297)

According to TEPCO, visual observations and a video recording made during the March 16, 2011, helicopter overflight (see Sidebar 2.1) showed that the Unit 4 pool contained water above the top of the fuel racks. However, TEPCO was unable to draw a similar conclusion about water levels in the Unit 3 pool because it was obscured by debris. Consequently, plant operators shifted their attention to the Unit 3 pool from March 16 to March 20.

The extensive visible damage to the Unit 4 reactor building and high level of decay heat in the Unit 4 pool continued to drive concerns about pool water levels. Operators began to add water to the Unit 4 pool starting on March 20, 2011. They first attempted to add water to the pool using water cannons and fire truck sprays on March 20 and March 22, respectively. These methods had limited effectiveness.[19] Operators then used a concrete pump truck to add water starting on March 22. In general, the effectiveness of this approach increased with time as personnel gained experience in remote operation of the truck.

TEPCO began systematic water-level measurements after April 12, 2011, and had refilled the pool by April 28. TEPCO then deliberately stopped adding water to the pool until May 6 but continued to measure the water level every day. The measurements were compared with energy-balance computations to determine whether water was leaking out of the pool, through either the gate seal or the pool liner. No leaks were detected.

Operators explored other approaches for adding water to the Unit 4 pool. They considered placing a pump on the refueling deck to transfer water from the reactor well and dryer-separator pit (with a combined volume of about 1,500 m^3) to the pool. This idea was dismissed because of concerns about personnel safety. They also considered using the FPC system, but aerial photography indicated that the check valve had been damaged by the building explosion.

The concrete pump truck was replaced with two other temporary measures in June 2011: The first was a fire hose run up the side of the building from a pump attached to a filtered water tank. The hose nozzle was attached to the fuel handling machine and directed downward into

[19] See discussion in Chapter 7 of Gauntt et al. (2012) and TEPCO (2012a, Attachment 9-1).

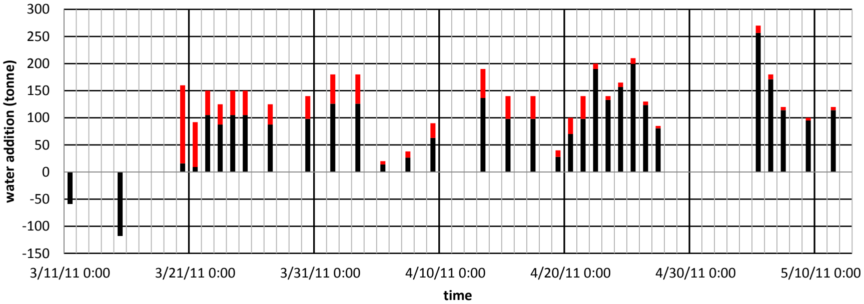

FIGURE 2.12 TEPCO-estimated water additions to the Unit 4 spent fuel pool between March 11 and May 12, 2011. Notes: The black bars represent TEPCO's best-estimate water additions presumably based on engineering judgment; the black bars plus red bars represent TEPCO's maximum-estimate water additions based on actual pumping flow rates and durations. The negative additions (i.e., water losses) shown on March 11 and March 15, 2011, represent losses due to earthquake- and explosion-induced sloshing, respectively. SOURCE: Data from TEPCO (2012a, Attachment 9-5).

the pool. The second was injection through the in-core neutron monitoring tubes at the bottom of the reactor to refill the reactor well and dryer-separator pit. These temporary measures were replaced after July 30 by the same system used to cool the Unit 2 and 3 pools.

TEPCO's estimates of water additions to the Unit 4 pool between March 11 and May 12, 2011, are shown in Figure 2.12. The black bars in the figure represent TEPCO's best-estimate water additions; the origin of these estimates is unclear.[20] The black bars plus red bars represent TEPCO's maximum-estimate additions; these were determined by measuring actual flow rates and flow durations. The "uncertainty" in water additions, represented by the red bars, is 940 tonnes, or 940 m^3 of water at standard pressure and temperature. This is approximately two-thirds the volume of the Unit 4 pool (see Table 2.1).

TEPCO used a mass- and energy-balance model to estimate water levels and temperatures in the Unit 4 spent fuel pool between March 11 and May 30, 2011. The results are shown in Figure 2.13. TEPCO speculates (TEPCO, 2012a, Attachments 9-1 and 9-5) that as water levels in the pool dropped because of evaporation, forces on the gates from water in the reactor well caused leakage around the gate seals, allowing additional water to enter the pool. This process is depicted in Figure 2.14. TEPCO also speculates that water leaked from the reactor well and dryer-separator pit into

[20] The committee could find no technical documentation for these estimates and presumes that they are based on engineering judgment.

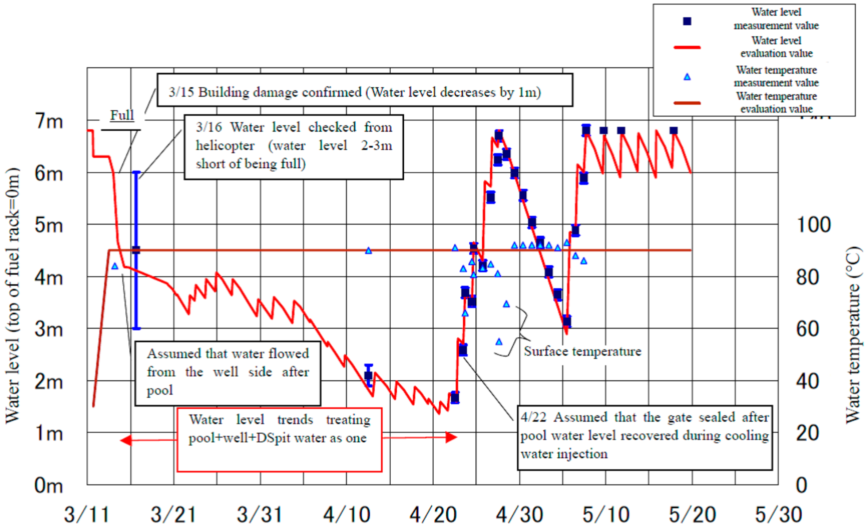

FIGURE 2.13 Water levels and temperatures in the Unit 4 spent fuel pool at the Fukushima Daiichi plant from March 11 to May 20, 2011. Notes: The squares and triangles are TEPCO-measured values of water level and temperature, respectively. The solid lines are TEPCO-modeled water levels and temperatures. SOURCE: TEPCO (2012a, Attachment 9-5, Figure 2).

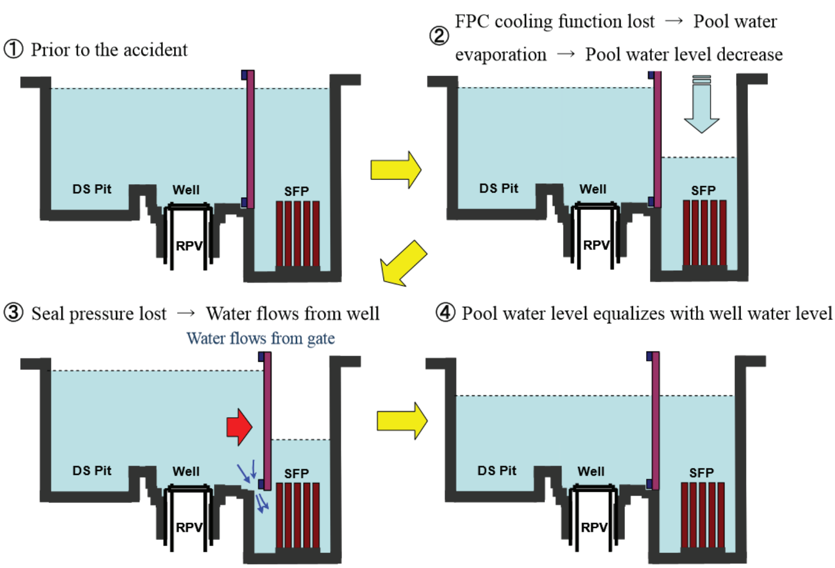

FIGURE 2.14 Sequence of events postulated by TEPCO for water inflow to the spent fuel pool from the reactor well. DS Pit = dryer-separator pit; RPV = reactor pressure vessel; SFP = spent fuel pool; FPC = fuel pool cooling and cleanup. SOURCE: TEPCO (2012a, Attachment 9-5, Figure 5).

the pool until April 22, 2011, when the gates were known to be secured. TEPCO estimates that water levels in the Unit 4 pool never declined below 1.5 m above the top of the fuel rack (Tateiwa, 2015) because of these water additions.

2.2.5 Damage to Spent Fuel in Units 1-4

There were significant concerns by TEPCO that the stored spent fuel in Units 1-4 may have been damaged by the earthquake and/or the explosion on March 15. TEPCO analyzed water samples obtained from the pools to determine whether radioactive contamination from damaged fuel was present. Measured activities and isotopic compositions are consistent with fallout from reactor core releases, with the possible exception of contributions from a small number of damaged spent fuel rods in Unit 2 (Jäckel, 2015). Evaluation of the potential contribution from damaged spent fuel is highly uncertain due to the masking effect of contamination due to fallout. Visual inspection of the fuel assemblies removed from the Unit 4 pool did not reveal any damage.

2.3 COMMITTEE ANALYSIS OF UNIT 4 POOL WATER LEVELS

TEPCO did not provide enough technical documentation for its water-level estimates in the Unit 4 spent fuel pool (Figure 2.13) to enable the committee to evaluate whether those estimates are realistic. In particular, there was insufficient information about the methods used to estimate water leakage around the gate seals and the total volume of leakage to the pool. These leakage estimates are critically important for obtaining realistic estimates of pool water levels.

The committee developed its own estimates of water levels in the Unit 4 pool between March 11 and May 12, 2011, to compare to the TEPCO estimates (Figure 2.13). The mathematical model and limitations are described in Appendix 2B, and the key features are as follows:

1. Evaporative water losses from the pool, which are driven by thermal heating of the pool water from radioactive decay in the stored spent fuel, were estimated using the steady-state energy-balance model described in Appendix 2B. Pool water level and temperature were assumed to be nominal (7 m above the fuel racks and 30°C, respectively) just prior to the March 11, 2011, earthquake and tsunami.
2. Water losses from the pool from sloshing were assumed to be 0.5 m for the March 11, 2011, earthquake and 1 m for the March 15 explosion, identical to TEPCO's estimates.

3. Water additions to the pool from external sources were assumed to be equal to TEPCO's best-estimate additions shown by the black bars in Figure 2.12.
4. Water additions to the pool from leakage around the gate seals were estimated using the orifice flow-rate correlation described in Appendix 2C. Water was assumed to flow in one direction from the reactor well to the spent fuel pool and only if the reactor water level was higher than the pool level; see Appendix 2C for further discussion of this issue.

The committee's estimates for the Unit 4 pool and reactor well and dryer-separator pit water levels are indicated by the blue and orange curves, respectively, in Figure 2.15. Note the following features in the committee's estimates:

- The vertical drop in water levels in the pool and reactor well and dryer-separator pit on March 11, 2011, was the result of earthquake-driven water sloshing as estimated by TEPCO.
- Following the earthquake, evaporative losses of water from the pool remained low until the pool temperature reached equilibrium,[21] at which point heat gained in the water from radioactive decay in the stored spent fuel balanced heat lost from water evaporation. The increase in pool-water evaporation rates is indicated by the change in the slope of the pool water-level curve starting around March 13, 2011.
- The explosion on March 15, 2011, caused an additional 1 m of water to slosh from the pool and reactor well and dryer-separator pit according to TEPCO.
- The model predicts that water began to leak from the reactor well into the pool on March 16, 2011. This leakage is indicated by the decrease in slope of the pool water-level curve (blue curve) and increase in slope of the reactor well and dryer-separator pit water-level curve (orange curve). Pool water levels are predicted to drop to within about 3.5 m above the top of active fuel[22] on March 22, 2011. Subsequently, the combination of water leakage around the gates and external water additions (Figure 2.12) was sufficient to maintain pool water levels above the top of the fuel.

[21] As shown in Table 2B.1 in Appendix 2B, equilibrium temperature in the Unit 4 pool is estimated to be 88°C, reached about 1.9 days after the FPC system shut down.
[22] TAF is 0.4 m below the top of the fuel racks.

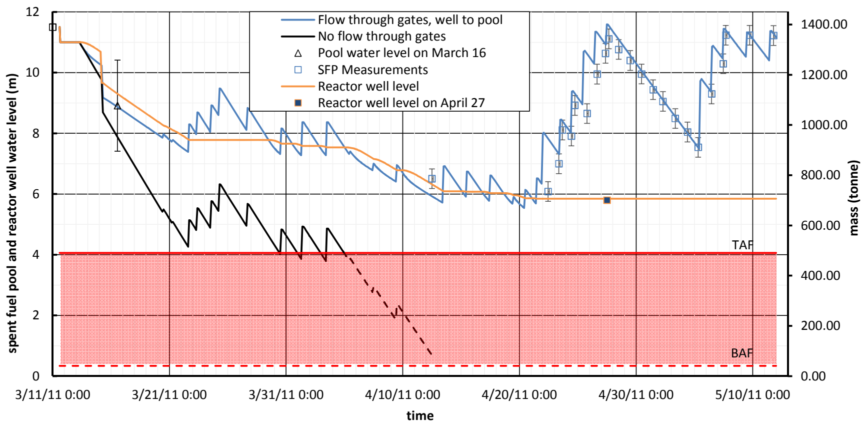

FIGURE 2.15 Committee-estimated water levels in the Unit 4 pool (blue curve) and reactor well (orange curve) using the models described in Appendixes 2B and 2C. Notes: The black curve shows a hypothetical scenario in which there is no water leakage from the reactor well and dryer-separator pit into the pool. The black square represents the water level in the reactor well measured by TEPCO on April 27, 2011, and the triangle is the TEPCO-estimated water level in the pool based on the helicopter overflight on March 16. TAF and BAF are top of active fuel and bottom of active fuel, respectively. The vertical scale in this figure is different from that in Figure 2.14. SOURCE: Committee-estimated water levels were estimated from model described in Appendixes 2B and 2C; water-level point data from Tateiwa (2015).

- Pool water levels dropped to less than 2 m above the tops of the racks on April 13 and again on April 20, 2011.[23] The water-level drop between April 4 and April 12 was a consequence of insufficient water injection amounts because TEPCO evaluated the need for additional water based on unreliable instrumentation for measuring water levels in the pool. Operators used a water-level indicator in the skimmer surge tank to determine whether the pool was full. They subsequently realized (TEPCO, 2012a, Attachment 9-5) that water oversprays onto the refueling deck were entering the skimmer surge tank through floor drains, bypassing the pool altogether, so less water was being added to the pool than estimated.[24]
- Leakage of water from the reactor well into the pool continued until April 21, 2011, when TEPCO was able to maintain sufficient

[23] TEPCO calculated a minimum height of 1.5 m above the top of the rack on April 20 (Tateiwa, 2015).

[24] After April 12, 2011, TEPCO was able to make direct visual observations of water injection using boom-mounted video cameras.

> water levels in the pool to keep the gates sealed. The committee estimates that between about 650 and 710 tonnes of water leaked from the reactor well into the pool between March 16 and April 21, equivalent to about half the volume of the pool.

Figure 2.16 compares the present committee's water-level estimates (blue curve in Figure 2.15) with water-level estimates from Oak Ridge National Laboratory (ORNL) (Wang et al., 2012) and water-level estimates and measurements from TEPCO (Figure 2.13). After April 22, 2011, there is good agreement among all of the water-level estimates and TEPCO's water-level measurements. This is consistent with water no longer leaking around the gate seals.

Prior to April 22, 2011, all of the water-level estimates have similar trends (Figure 2.16). However, the ORNL and committee estimates indicate that the pool experienced larger water-level excursions and lower absolute water levels compared with the TEPCO estimates. The ORNL and committee estimates indicate that water levels ranged between 1 and 2 m above the top of active fuel between April 10 and 22; TEPCO estimates that water levels never fell below 2 m above the top of active fuel. These differences are apparently due to TEPCO's assumption that the reactor well and spent fuel pool were "hydraulically connected" before April 22. This connection effectively doubles the mass of water available to absorb heat from the spent fuel and reduces changes in pool water levels.

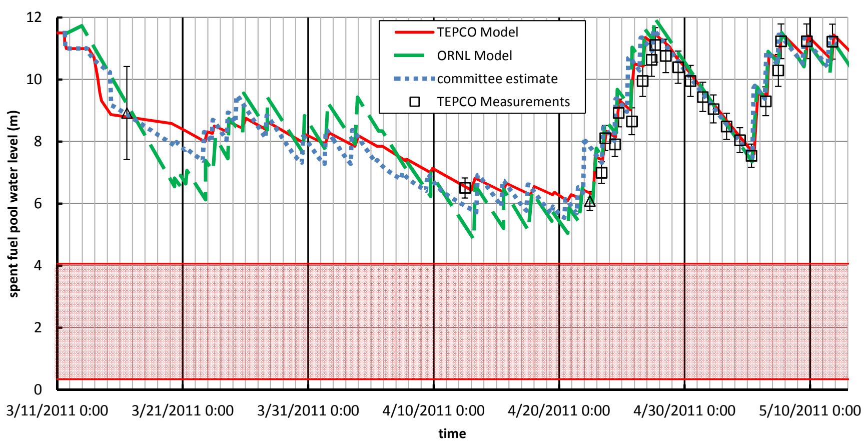

FIGURE 2.16 Comparison of Unit 4 spent fuel pool water-level estimates from TEPCO (2012a) (red curve), ORNL (Wang et al., 2012) (green curve), and the present committee (dotted blue curve). Note: Also shown are the measurements of pool water levels made by TEPCO starting on April 22 (black squares).

There are substantial uncertainties due to the simplified assumptions used in the model for the committee's water-level estimates between March 11 and April 20, 2011, because

1. Water losses from earthquake- and explosion-related sloshing were not measured. The estimates were likely based on engineering judgment.
2. The committee's estimates for water leakage from the reactor well and dryer-separator pit to the pool prior to April 20, 2011—between about 650 and 710 tonnes—are based on an engineering model for gate leakage that has large uncertainties in the parameters.
3. TEPCO's effectiveness estimates for water additions to the pool (Figure 2.12) also appear to be based on engineering judgment and are subject to significant uncertainty: at least 940 tonnes, the sum of the red bars in Figure 2.12.
4. Pool water levels prior to April 12, 2011, were not measured. As noted previously, the water-level indicator in the skimmer surge tank provided misleading information on pool water levels prior to this date.
5. The committee's steady-state energy-balance model does not account for the presence of fuel or racks other than through the decay heat of the spent fuel. A number of effects not accounted for in the model become important once the water level drops below the top of the racks, most notably the reduction in cross-sectional water area. Consequently, the water-level estimate without gate leakage shown in Figure 2.15 is indicated by a dashed line once it falls below top of the racks after April 5. This estimate is increasingly unreliable below this level because of geometrical inaccuracy and also because of other physical phenomena not accounted for by the model, including rack and fuel heat capacity, multiphase flow, film boiling, cladding oxidation, flow blockage, and change in geometry with loss of cladding integrity. Computations that include many of these effects are discussed in the Sandia analysis of a hypothetical loss-of-cooling accident in Unit 4 (see Chapter 8 of Gauntt et al., 2012).

Two important observations can be made from the committee's analysis of water levels in the Unit 4 pool. First, because of the substantial uncertainties cited above, the committee cannot rule out the possibility that spent fuel in the Unit 4 pool became partially uncovered sometime prior to April 21, 2011. If the fuel was uncovered, however, then it was not substantial enough to cause fuel damage or substantially increase external dose rates in areas

TABLE 2.2 Estimated Peak Radiation Dose on the Refueling Deck Next to a Pool with Recently Offloaded Spent Fuel as a Function of Pool Water Level

Height of Water Level above Racks (m)	Peak Dose Rate on the Refueling Deck
0.0	450-600 rem/hr
0.6	>25 rem/hr
1.0	1.6-1.7 rem/hr
3.0	< 0.1 mrem/hr

SOURCE: USNRC (2014a, pp. 88-89).

around the Unit 4 building. Fuel damage will not begin immediately when the water level drops below the top of the rack. Simulations of loss-of-cooling accidents (Gauntt et al., 2012) predict that it is possible to recover without fuel damage as long as the collapsed[25] water level does not drop below the midheight of the fuel for an extended period of time.

Second, leakage through the gate seals was essential for keeping the fuel in the Unit 4 pool covered with water. Had there been no water in the reactor well, there could well have been severe damage to the stored fuel and substantial releases of radioactive material to the environment. This is the "worst-case scenario" envisioned by then–Atomic Energy Commission of Japan Chairman Dr. Shunsuke Kondo (see Sidebar 2.3).

To illustrate this second observation, the committee modeled a *hypothetical scenario* in which there is no water leakage into the Unit 4 pool from the reactor well and dryer-separator pit. The results are shown by the black curve in Figure 2.15. Without water leakage, pool water levels could have dropped well below the top of active fuel (located 4 m above the bottom of the pool) in early April 2011.

In the committee's judgment, the events in the Unit 4 pool should serve as a wake-up call to nuclear plant operators and regulators about the critical importance of having robust and redundant means to measure, maintain, and, when necessary, restore pool cooling.

The events in the Unit 4 pool also have important implications for accident response actions. As water levels decrease below about 1 m above the top of the fuel racks, radiation levels on the refueling deck and surrounding areas will increase substantially, limiting personnel access (Table 2.2). Moreover, once water levels reach approximately 50 percent of the fuel

[25] Collapsed water level refers to a computed height of liquid water that has the same mass as the multiphase mixture (liquid, gas, and vapor) of water that is covering the fuel.

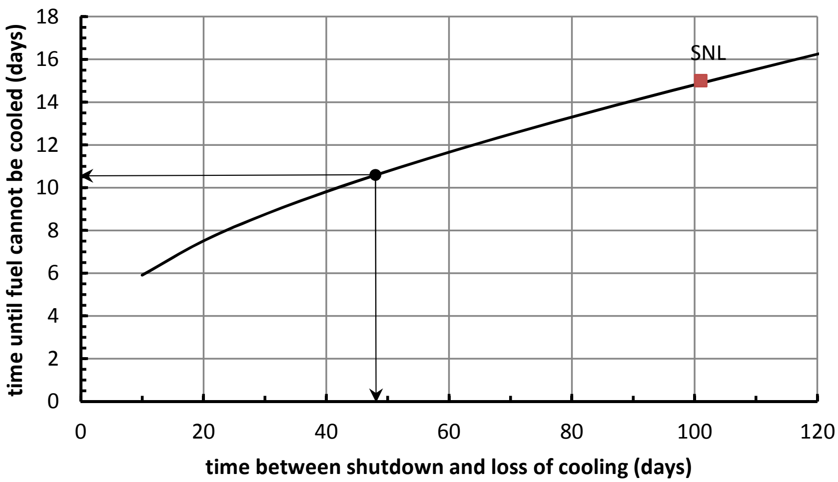

FIGURE 2.17 The curve indicates the time until water levels in the Unit 4 pool decline to 50 percent of fuel height and can no longer be cooled by evaporation of pool water (y axis) as a function of elapsed time between reactor shutdown/core offload and loss of pool cooling (x axis). Note: This estimate was developed using the model in Appendix 2B with the same assumptions about the fuel and Unit 4 building condition as Scenario 2 of Gauntt et al. (2012); the red square labeled "SNL" is the result of the Sandia's MELCOR simulation for that scenario. The black filled circle indicates a hypothetical situation in which, if the fuel had been offloaded only 48 days before the accident, the first water additions to the Unit 4 pool would have taken place just as the water level was reaching the midheight of fuel.

assembly height, the tops of the rods will begin to degrade, changing the fuel geometry and increasing the potential for large radioactive material releases into the environment (Gauntt et al., 2012, p. 183).

These observations bear directly on the safety of pool storage following large offloads of fuel from reactors. To illustrate this point, consider what might have occurred in the Unit 4 spent fuel pool had the reactor been shut down and the core been offloaded to the pool about 2 months later (i.e., 48 days before March 11) than it was (i.e., 102 days) and had there been no water leakage from the reactor well and dryer-separator pit. The committee estimates that pool water levels would have reached 50 percent of fuel assembly height (Figure 2.17) before 10.6 days had elapsed—which was the time elapsed between the onset of the accident on March 11 and the first addition of water to the pool in Unit 4. In this hypothetical situation, if the core had been offloaded closer to the time of the accident or if the water addition had been delayed longer than 10.6 days, then there could have been damage to the fuel with the potential for a large release of radioactive material from the pool, particularly because the most recently offloaded (and highest-power) fuel was not dispersed in the pool but was concentrated in adjacent locations within the racks (see Figure 2.10).

2.4 FINDING AND RECOMMENDATION

FINDING 2.1: The spent fuel storage facilities (pools and dry casks) at the Fukushima Daiichi plant maintained their containment functions during and after the March 11, 2011, earthquake and tsunami. However, explosions in the Unit 1, 3, and 4 reactor buildings damaged spent fuel handling facilities and equipment, introduced heavy debris into the pools, and provided enhanced pathways for releases of radioactive materials from the damaged reactors into the environment. These events hindered efforts by plant operators to monitor conditions in the pools and restore critical pool-cooling functions. The lack of reliable real-time information about the pools created substantial difficulties in responding to the accident and led to increased public anxiety. Nevertheless, plant personnel were able to improvise and provide needed cooling to avoid pool uncovery and potential radiological consequences. The leakage of water into the Unit 4 pool from the reactor well/dryer-separator pit was a key factor for determining its water level and may have prevented fuel uncovery before plant personnel were able to add water.

RECOMMENDATION 2.1: The U.S. nuclear industry and its regulator should give additional attention to improving the ability of plant operators to measure real-time conditions in spent fuel pools and maintain adequate cooling of stored spent fuel during severe accidents and terrorist attacks. These improvements should go beyond the current post-Fukushima response to include hardened and redundant (1) physical surveillance systems (e.g., cameras), (2) radiation monitors, (3) pool temperature monitors, (4) pool water-level monitors, and (5) means to deliver pool makeup water or sprays even when physical access to the pools is limited by facility damage or high radiation levels.

2.4.1 Discussion

The spent fuel pools in Units 1-4 were sufficiently robust to survive the earthquake and explosions, although the support structure beneath the Unit 4 pool needed to be strengthened. However, the spent fuel pool gates in Unit 3 were damaged during the accident and the pool gates in Unit 4 appear to have leaked water. Although debris entered the pools and covered portions of the racks supporting the fuel assemblies in Unit 3, the fuel does not appear to have been damaged with the possible exception of a small number of fuel rods in the Unit 2 pool.

Plant operators had not planned for or trained to respond to the conditions that existed in the Unit 1-4 spent fuel pools following the March 11, 2011, earthquake and tsunami:

1. Primary and backup pool cooling systems had failed because of the loss of all power. There were no plans or equipment available for adding emergency makeup water or implementing alternate cooling strategies.
2. Water-level and temperature monitoring instrumentation had also failed because of the loss of power. The limited range of monitoring instrumentation greatly reduced its effectiveness even after power was restored.
3. Explosions in Units 1, 3, and 4 damaged the reactor buildings, introduced debris onto the refueling decks and into the spent fuel pools, and hindered visual observations of pool conditions.
4. Radiological conditions hindered access to areas around the buildings and limited personnel access to the refueling decks and pools.

Operators successfully developed and deployed improvised means to monitor and cool the pools using helicopters, fire trucks, water cannons, concrete pump trucks, and ad hoc connections to installed cooling systems. These operator actions were clearly essential for preventing damage to the stored spent fuel and release of radioactive materials to the environment.

The committee's Recommendation 2.1 calls on the U.S. nuclear industry and its regulator to improve the ability of plant operators to monitor real-time conditions in spent fuel pools and maintain adequate cooling of stored spent fuel during severe accidents or terrorist attacks. The recommended improvements in plant monitoring systems include the following:

- Remote surveillance of pools and refueling decks,
- Radiation levels on the refueling deck,
- Pool temperatures, and
- Pool water levels.

The lack of reliable information on pool water levels and temperatures at the Fukushima Daiichi plant created unnecessary anxiety about the condition of the stored spent fuel and may have also created false priorities for allocating resources. Reliable and hardened instrumentation is just as necessary for the spent fuel pools as it is for the reactor safety systems.

Some of the committee-recommended improvements in NRC (2014) are already being made by the U.S. Nuclear Regulatory Commission (USNRC) and nuclear industry. The USNRC required its licensees to upgrade water-level instrumentation in their spent fuel pools following the Fukushima accident.[26] The regulatory guidance is to provide two independent instrumentation systems, primary and backup, with at least a 1-ft (0.3-m) resolution down to 10 ft (3 m) above the top of the racks and 3.5-ft (1-m) resolution from the 10-ft (3-m) level to the top of the racks. The systems must be powered from independent power sources and have provisions for attaching power sources independent of plant AC and DC power distribution systems, for example, portable generators or batteries. The systems must be seismically rugged, must be operable under severe accident conditions,[27] and must be installed so that they have reasonable protection in case of damage to the structure over the pool. They must also be designed so that spent fuel pool water levels can be read from the control room, an alternate shutdown panel, or other accessible locations.

It is essential to have the capability to monitor water-level trends in the pool when they are within 3 m of the top of the fuel racks. This monitoring capability is needed to ensure that water additions are effective and that water levels are adequate to shield radiation from the stored spent fuel. Radiation monitors available on the refueling deck could serve as a proxy measure of water levels when they are close to the top of the fuel racks (see Table 2.2). However, these monitors may not be operable during a severe accident, and their readings may be difficult to interpret if the refueling deck is contaminated with radioactive material.

Some of the committee-recommended improvements have not been made by the USNRC or nuclear industry. In particular, the USNRC has not required plant licensees to install pool temperature monitors. *In the committee's judgment, pool temperature measurements are essential in an accident to evaluate independently whether drops in pool water levels are due to evaporation or leakage.*[28] As noted in Chapter 5 of the committee's phase 1 report (NRC, 2014), the Advisory Committee on Reactor Safeguards (ACRS, 2012a, p. 5) has also recommended the installation of pool-temperature monitoring instrumentation that would display directly in the main control room. ACRS (2012b) also recommended that higher water-level spatial resolution as well as temperature measurements were needed to enable operators to respond in a timely and appropriate fashion to spent fuel pool accidents.

[26] The specifications are given in USNRC guidance JLD-ISG-2012-03.

[27] This water-level instrumentation must survive down to TAF.

[28] Like the water-level instrumentation described previously, the temperature instrumentation must have independent power, be seismically rugged, and operate under severe accident conditions.

The committee discussed pool-water temperature monitoring in its phase 1 report (see NRC, 2014, pp. 162-163) and recommended a systematic evaluation be undertaken to determine if such monitoring was needed at U.S. nuclear plants. The committee commented that once this evaluation was completed,

> "nuclear plant licensees and the USNRC might conclude that additional temperature sensors should be placed in pools to provide confirmatory information about the thermodynamic state of water inventories."

The committee again encourages the U.S. nuclear industry and the USNRC to complete this evaluation expeditiously.

The U.S. nuclear industry is already making good progress in improving the ability of plant operators to maintain adequate cooling of stored spent fuel during severe accidents or terrorist attacks:

- Under its B.5.b initiative,[29] the industry has pre-positioned equipment and developed procedures to add makeup water to spent fuel pools and cool the stored fuel assemblies with water sprays (see Chapter 3).
- Under its Diverse and Flexible Coping Strategies (FLEX) initiative[30] (NEI, 2012), the industry has pre-positioned equipment and developed procedures to provide water and power to address the USNRC's Mitigation Strategies Order (USNRC, 2012a).
- Both of these initiatives were discussed in the committee's phase 1 report (NRC, 2014, see especially Appendixes F and G).

These initiatives may employ separate sets of equipment and procedures.[31]

- The industry's FLEX strategy (NEI, 2012, p. 11) specifies that "FLEX mitigation equipment should be stored in a location or locations such that it is reasonably protected such that no one external event can reasonably fail the site FLEX capability. Reasonable protection can be provided for example, through provision of multiple sets of portable on-site equipment stored in diverse locations or

[29] Codified in 10 CFR 50.54(hh)(2) for responding to large fires and explosions. These requirements are discussed in the committee's phase 1 report (NRC, 2014). See especially Appendix G.

[30] This initiative is also discussed in the committee's phase 1 report (NRC, 2014). See especially Appendix F.

[31] The FLEX strategy allows for the use of B.5.b equipment if it meets applicable FLEX requirements (NEI, 2012, p. 53).

through storage in structures designed to reasonably protect from applicable external events" (NEI, 2012, p. 11).

- B.5.b equipment, on the other hand, may not be protected from the kinds of external events for which FLEX equipment is designed to mitigate. Consequently, it might not be readily accessible to plant personnel during an external event that limited physical access to the pools. Moreover, this equipment is not necessarily compatible with the FLEX equipment.

Better coordination and integration of the B.5.b and FLEX equipment and procedures could enhance plant operators' ability to respond to severe accidents and terrorist attacks. It could also provide cost savings and efficiency improvements.

The capabilities provided by B.5.b and FLEX provide for spent fuel pool mitigation strategies of 200-gpm (gallon per minute) water spray and 500-gpm makeup water for at least 12 hours. These are the types of capability that had to be improvised at Fukushima Daiichi using fire engines and concrete pump trucks. The spray solution of FLEX[32] anticipates the use of hoses on the refueling deck; this is problematic if the decks are severely damaged or contaminated. Solutions that avoid these problems include the concrete pump trucks employed at the Fukushima Daiichi plant or pre-positioned nozzles (protected from falling debris) that can be remotely operated using water connections external to the spent fuel enclosure or reactor building.

Finally, the damage observed in the Unit 3 gates (Figure 2.9) demonstrates a pathway by which a severe accident could compromise spent fuel pool storage safety: drainage of water from a spent fuel pool through a damaged gate breach into an empty volume such as a dry reactor well or fuel transfer canal. A gate breach could drain a spent fuel pool to just above the level of the racks in a matter of hours, and the resulting high radiation fields on the refueling deck (Table 2.2) could hinder operator response actions. The committee judges that an effort is needed to assess the containment performance of spent fuel pool gates under severe accident conditions during all phases of the operating cycle.[33] Such an assessment could be

[32] USNRC (2016) notes that the FLEX portable spray capability (utilizing portable spray nozzles from the refueling floor with portable pumps) is not required when a pool is located below grade or when a seismic hazard analysis shows that the pool will maintain its integrity.

[33] Assessment of spent fuel pool performance, including gate leakage, is not a new topic for the USNRC. A review of historical data in 1997 (USNRC, 1997c) documented numerous instances of significant accidental drainage of pools in pressurized water reactor and BWR plants due to various failures including gate seals. The report recommended that "[t]he overall conclusions are that the typical plant may need improvements in SFP [spent fuel pool] instrumentation, operator procedures and training, and configuration control" (p. xi). Furthermore,

carried out as part of the committee-recommended risk assessment in its phase 1 report (NRC, 2014) and in the risk assessments recommended in Chapters 4 and 5 of this report.

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the report goes on to identify the most prevalent reason for loss of pool inventory was leaking fuel pool gates. Given the potential for gate leakage under normal operations it is not surprising that it is also an issue under severe accident conditions.

APPENDIX 2A
Spent Fuel Storage in Units 5 and 6, the Common Pool, and Dry Cask Storage at the Fukushima Daiichi Plant

This appendix provides information about spent fuel storage in Units 5 and 6, the common pool, and dry cask storage at the Fukushima Daiichi plant. Table 2.1 in the main body of the chapter provides information about spent fuel storage quantities.

2A.1 UNITS 5 AND 6

The Unit 5 pool lost cooling after AC power and pumps for circulating seawater to the heat exchanges were knocked out by the tsunami. The pool temperature reached about 69°C on March 18, 2011, when improvised sea water pumps were put into operation. The residual heat removal system was used to alternately cool the reactor and the pool, during which time pool temperatures fluctuated between 30°C and 50°C. After June 25, normal cooling was reestablished with the FPC system, allowing pool temperatures to be maintained at around 30°C.

Seawater pumps were also lost in Unit 6 after the earthquake and tsunami. However, one of the emergency diesel generators continued to function. The pool temperature reached a maximum of about 68°C before the sea water pumps were restored on March 19, 2011. The residual heat removal system was used to alternately cool the reactor and the pool, during which time pool temperatures fluctuated between 20°C and 40°C.

2A.2 COMMON SPENT FUEL POOL

The earthquake and tsunami knocked out power for the FPC system, causing pool water temperatures to rise to about 73°C before the power was restored on March 24 (TEPCO, 2012a, p. 300).

2A.3 CASK STORAGE

Casks have been in use at the Fukushima Daiichi plant since 1995. The casks are steel, equipped with an inner and outer bolted closures that can be removed for inspection, and bolted to the foundation of the cask storage building, which is located at a low elevation close to the quay (see Figure 2.1). Nine casks containing a total of 408 fuel assemblies were in storage on March 11, 2011 (TEPCO, 2012a, Attachment 9-9).

The building lost power and was inundated with sea water, sand, and debris by the tsunami, and the doors and louvers ventilating the building were damaged. However, the casks were not damaged or displaced, and air flows were not significantly obstructed (TEPCO, 2012a, p. 300). Inspection of the cask interiors in March through May 2013 revealed that there was no leakage of seawater into or helium out of the casks, and there was no damage to the fuel bundles or baskets within the casks (Tateiwa, 2015; Wataru, 2014).

APPENDIX 2B
Analysis of Spent Fuel Pool Heat-up

This appendix describes the committee's steady-state energy-balance model that is used to estimate water losses in the Unit 1-4 spent fuel pools. For more details about modeling spent fuel pools, see the discussion in EPRI (2012a), Wang et al. (2012), and Gauntt et al. (2012). The key assumptions and limitations of the committee's model are the following:

- The model is based on the conservation of mass and energy using the fundamental principles of thermodynamics applied to a control volume surrounding the pool water.
- The pool is well mixed and can be characterized by a single temperature.
- The effect of spent fuel is only as a thermal energy source characterized by decay heat.
- There is a transient period of pool heating during which vaporization is neglected.
- Once the transient period is over, vaporization of water and thermal energy loss balances the thermal energy addition due to the spent fuel decay heat. This energy balance keeps the pool temperature constant in time and is treated as a steady-state condition although the water level may vary because of water loss or addition.
- The thermal energy losses due to radiation, convection, and conduction are characterized by a single value for each pool.
- The reduction in cross-sectional water area within the rack regions is not treated.
- Fuel cladding or rack oxidation, multiphase flow, and changes in geometry of the fuel or racks are not treated.
- The specific heat, density, and heat of vaporization of water are assumed to have constant values.
- Water addition is at a constant temperature of 10^{o}C, and the initial pool temperature is 30^{o}C.
- Water evaporated from the pool is completely dispersed into the atmosphere and does not condense on the remaining structure and flow back into the pool.

As a consequence of these simplifying assumptions, the model results are limited in applicability to pool water levels above the top of the fuel racks, and the quantitative results have an associated uncertainty that the committee has not characterized in detail.

The values of key inputs such as pool sizes, decay heat, thermal losses, and water addition amounts have been taken from TEPCO (2012a), and the water properties are average values based on actual thermophysical data. The model results have been compared with those of other investigators for Unit 4 and found to be in reasonable agreement (Figure 2.16). The main contribution of the committee's model was to examine in detail the role of leakage through the gates between the reactor well and the spent fuel pool in Unit 4, a key issue that was identified during the course of the committee's deliberations.

Following the loss of active cooling in a pool, the thermal power (decay heat) $\dot{Q}_{sf}$ from the stored spent fuel heats the water and changes its buoyancy, which creates circulation within the pool. Water rises through the assemblies to the surface of the pool where some of it evaporates, cooling the water at the surface. This cooler water sinks into the pool. Some of it mixes with the rising warmer water in the central portion of the pool, and some of it sinks to the bottom along the sides and unracked parts of the pool, travels under the racks, and is entrained into the flow produced by the rising hot fluid inside the fuel assemblies (Figure 2B.1).

Assuming that the mass M of water within the pool is well mixed and at common temperature T, the conservation of energy in the pool can approximately modeled as

$$MC_p \frac{dT}{dt} = \dot{Q}_{sf} - \dot{Q}_{loss} - \Delta H_{fg} \dot{M}_{vap} - C_p \Delta T \dot{M}_{add},$$

where C_p is the water specific heat capacity, ΔH_{fg} is the heat of vaporization, $\dot{M}_{vap}$ is the rate of vaporization from the pool, ΔT is the difference between the pool and added water temperature, and $\dot{M}_{add}$ is the rate at which water is being added to the pool. The thermal power (decay heat) $\dot{Q}_{sf}$ of the spent fuel is balanced primarily by the energy loss due to vaporization as represented by the term $\Delta H_{fg} \dot{M}_{vap}$. There is a much smaller (about 10 percent of the decay heat in Unit 4) loss of energy, $\dot{Q}_{loss}$, through radiation and convection from the pool upper surface and conduction to the pool liner and concrete; these losses were estimated by EPRI (2012a) and TEPCO (2012a, Attachment 9-1) and are modest but not negligible. The committee has used the values[1] proposed by TEPCO for the purposes of estimating approximate times for heat-up and evaporation of spent fuel pools.

The conservation of mass can be modeled as

$$\frac{dM}{dt} = \dot{M}_{add} - \dot{M}_{vap} - \dot{M}_{leak}.$$

[1] Unit 1, 0.08 MW; Unit 2, 0.11 MW; Unit 3, 0.11 MW; Unit 4, 0.16 MW. The heat loss rates for Unit 5, Unit 6, and the common pools were not given by TEPCO; the committee estimated a value of 0.11 MW for Units 5 and 6, and 0.33 MW for the common pool.

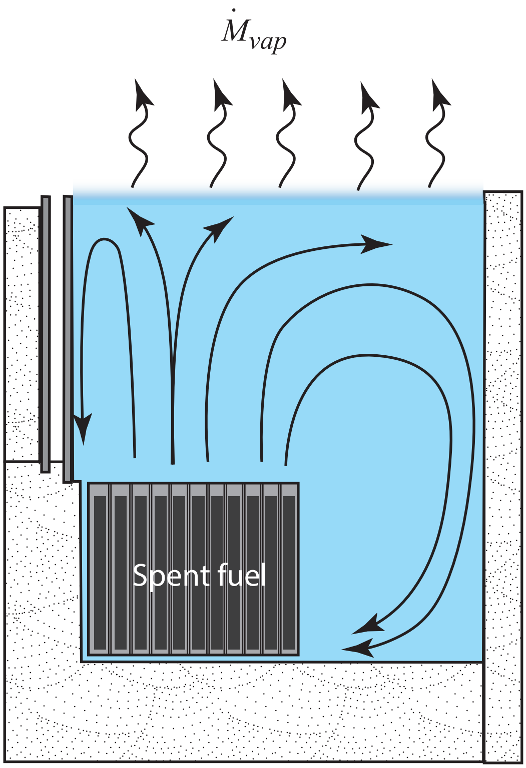

FIGURE 2B.1 Schematic of spent fuel pool showing the circulation pattern created by the buoyant plume of hot water above the spent fuel.

The rate of vaporization, $\dot{M}_{vap}$, depends on the mass transfer rate across the layer of water vapor (steam) in the air above the surface of the pool (Figure 2B.2). The vaporization creates a humid atmosphere above the pool if the reactor building is intact. This will reduce the vaporization rate for a given pool temperature, and, if the vaporization rate is fixed, this will increase the pool temperature. Additionally, the evaporated water will condense on the interior of the cold reactor building with some fraction returning to the pool. In certain MELCOR simulations, the net effect is to decrease the rate at which the pool water level decreases by up to a factor of 2 over the situation in Unit 4 where the building structure was demolished (Gauntt et al., 2012, p. 192).

The rate of water addition to the pool, $\dot{M}_{add}$, is determined by the effectiveness of mitigation measures and possible infiltration from the reactor well through the gates (see Appendix 2C), offset by the rate of water

FIGURE 2B.2 Water evaporating from the surface of the Unit 4 spent fuel pool. SOURCE: TEPCO image released June 29, 2011.

leakage, $\dot{M}_{leak}$, from the pool, which depends on the size and location of the leak. Added water will typically be cooler than that in the pool, so some of the decay heat is used to raise the temperature of the water by $\Delta T = T - T_{add}$. In the simplest implementation of the pool energy-balance model, the water specific heat capacity (C_p) is assumed to be constant at 4.184 kJ/kg K and the heat of vaporization is also assumed to be constant, ΔH_{fg} = 2.3 MJ/kg. At Fukushima, the added water temperature was about 10°C, and the initial pool temperatures were about 30°C before the loss of pool cooling.

The effect of a humid atmosphere on water evaporation from the pool can be predicted by a simple engineering model for the mass transfer based on a correlation for the mass transfer coefficient h_m to find the evaporation rate through the water-air surface area A:

$$\dot{M}_{vap} = Ah_m\,[\rho_{sat}(T) - \rho_{sat}(T_{air})],$$

where $\rho_{sat}(T)$ is the mass density of the saturated water vapor at temperature T.

The mass transfer coefficient can be estimated as

$$h_m = D\frac{Sh}{W},$$

where *Sh* is the Sherwood number, *D* is the mass diffusivity of water vapor in air, and *W* is the characteristic dimension of the pool surface, in this case the width. The Sherwood number depends on the conditions in the mixed water vapor–air layer above the pool, particularly the air temperature, relative humidity, and air circulation rates within the building or wind conditions for an exposed pool; see the discussion by Wang et al. (2012) and Hugo and Omberg (2015).

The flow within the water vapor–air layer about the pool is turbulent; it is difficult to accurately predict the vaporization rates with certainty even if the conditions above the pool are known or can be reliably estimated. As a consequence, vaporization rates estimated by mass transfer correlations are highly uncertain (Hugo and Omberg, 2015). Fortunately, the quasi-steady-state energy-balance method does not rely on the mass transfer computation, and reliable predictions of vaporization rates can be made as long as the pool temperature and thermal radiation losses can be estimated or are known from measurements.

Without mitigation, the pool will heat up until a quasi-steady-state equilibrium condition is reached in which the rate of energy loss due to vaporization balances the thermal power input from the spent fuel, $\dot{Q}_{sf}$, correcting for the heat lost by radiation, conduction, and convection with the term $\dot{Q}_{loss}$, which is estimated using engineering correlations[2] for heat transfer:

$$\dot{Q}' = \dot{Q}_{sf} - \dot{Q}_{loss} = \Delta H_{fg}\dot{M}_{vap}.$$

Values of the effective thermal energy input rate $\dot{Q}'$ are given in column 3 of Table 2B.1. In the range from 30°C to 90°C, the heat of vaporization, ΔH_{fg}, varies only slightly[3] with temperature so that an estimate of the vaporization rate can be obtained using the nominal value of 2.3 MJ/kg. At the steady-state condition, termed *thermal equilibrium*, the temperature of

[2] The value of the effective heat transfer coefficient from the surface will vary with pool temperature but the variation is modest, ranging from 10.4 W/m^2 K for the Unit 1 pool up to 12.4 W/m^2 K for the Unit 4 pool. TEPCO used a value of 11.6 W/m^2 K in its estimates. Conduction heat transfer is transient and the rate of energy loss will decrease with increasing time.

[3] The enthalpy of vaporization varies from 2,430 to 2,282 kJ/kg K, a variation of about 6 percent over this range.

TABLE 2B.1 Model Estimates for the Spent Fuel Pools at the Fukushima Daiichi Plant

Unit	M_o (tonne)	$\dot{Q}'^{a}$ (MW)	$T_{eq}{}^{b}$ (°C)	t_{heat} (day)	$\dot{M}_{evap}$ (tonne/day)	t_{mid} (day)
1	990	0.10	55	12	4.0	229
2	1383	0.51	70	5.3	19	63
3	1383	0.43	68	5.9	16	74
4	1383	2.10	88	1.9	79	16
5	1383	0.90	69	2.9	34	35
6	1453	0.76	68	3.5	29	44
Common	3716	0.77	73	10	29	110

[a] The values shown are the effective thermal energy input rate (decay heat minus thermal energy losses due to heat transfer) and are lower than the decay heat values given in Table 2.1.
[b] Values for Units 1-4 based on estimates from Wang et al. (2012). Other values based on maximum measured temperatures.

the pool attains a maximum value consistent with a vaporization rate that results in removing energy at a rate that exactly balances the decay heat generated by the spent fuel minus the heat transfer losses.

The result of the committee's energy balance and vaporization rate computation[4] for the Unit 4 pool is shown in Figure 2B.3. Equilibrium temperatures for the Fukushima Daiichi pools are predicted to be between 55°C and 88°C depending on the particular pool; this range is comparable to the actual measured values. The equilibrium temperature for the Unit 4 pool is predicted by the committee's model to be 87°C, and the vaporization rate is about 0.9 kg/s (78 tonne/day). The measured pool temperature was about 84°C on March 14, 2011, and about 90°C on April 12, 2011.

2B.1 MODEL RESULTS FOR FUKUSHIMA POOLS

The committee applied a simplified version of this model to the initial stage of a Fukushima-like scenario of loss of pool cooling without mitigation or leakage. The energy addition to pool water is divided into two

[4] The committee's computation used the "stagnation film boundary layer" mass transfer analysis accounting for induced convection and the significant mass fraction of water in the region above the pool surface; see the discussion by Lienhard and Lienhard (2015, Section 11.8). The computational result shown in Figure 2B.3 accounted for the temperature dependence of all thermophysical properties and is similar in concept to that used by Wang et al. (2012) but included conventional engineering models of the convective, radiative, and conduction energy losses as well as evaporative losses.

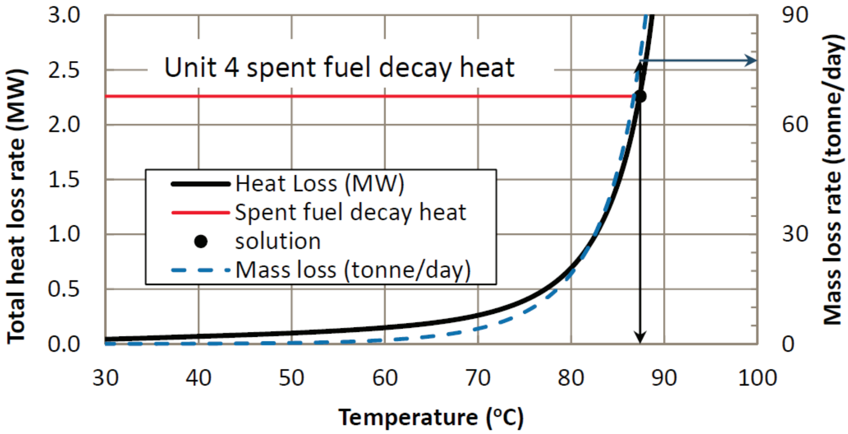

FIGURE 2B.3 Committee model for total heat loss rate and evaporative mass loss rate as a function of pool water temperature for Unit 4 (also valid for Units 2, 3, and 5). Notes: The intersection of the spent fuel decay heat line (red) with the total heat loss rate curve (black) is the graphical solution (filled circle) to the energy balance and determines the equilibrium temperature of 87°C given by the vertical arrow and the mass loss rate curve (dashed blue) of 78 tonnes/day (0.9 kg/s) given by the horizontal arrow. The predicted equilibrium temperature is comparable to actual measured values and those predicted by Wang et al. (2012).

steps. First, pool water heats up until the surface reaches the equilibrium temperature with negligible loss of mass:

$$M_o C_p \frac{dT}{dt} \approx \dot{Q}'.$$

Second, the water in the pool remains at constant temperature and evaporates:

$$\dot{Q}' \approx \Delta H_{fg} \dot{M}_{vap}.$$

Based on these simplifications, the time to heat a pool of mass M_o to the equilibrium temperature T_{eq} from the initial temperature T_0 is given by

$$t_{heat} = \frac{M_o C_p \left(T_{\mathrm{eq}} - T_0\right)}{\dot{Q}'}.$$

The mass of the pool at a time t relative to the end of the heating period will be

$$M = M_o - \frac{\dot{Q}'}{\Delta H_{fg}} t.$$

This estimate is reasonable until the water level approaches the top of active fuel and a two-phase (water vapor and liquid) mixture is created due to boiling in water surrounding the spent fuel. Evaporative cooling will become progressively less effective as water continues to be vaporized and the channels of the fuel assemblies are filled with steam or a two-phase mixture rather than liquid water. Simulations by Sandia using MELCOR (Gauntt et al., 2012, p. 192) indicate that, for the conditions of the Unit 4 pool, the assumption of a constant vaporization rate appears to be approximately valid up until the collapsed water level reaches the midplane (2.2 m above the pool floor) of the active portion of the fuel assembly. A comparison between the committee model and the MELCOR results for the case of a full pool with no building (Gauntt et al., 2012, Figure 118) is shown in Figure 2B.4.

The comparison in Figure 2B.4 demonstrates that the simplified model of the committee is in reasonable agreement with the MELCOR model and overpredicts the time to reach the fuel midplane by 6 to 13 percent (1 to 2 days) depending on the assumptions about energy loss. This over-prediction is reasonable given that the committee's model does not account for the reduction in water cross-sectional area within the racks and other phenomena such as fuel temperature increase and heat transfer into the racks. With this validation of the modeling approach, the time t_{mid} required for the collapsed water level to reach the critical height corresponding to the midplane of the fuel for all pools can be estimated by computing the time needed to evaporate the water ΔM_{mid} above this elevation:

$$t_{mid} = \frac{\Delta M_{mid} \Delta H_{fg}}{\dot{Q}'}.$$

The results are given in the last column of Table 2B.1. The amount of water loss, ΔM_{mid}, corresponding to a reduction in pool collapsed water level to the midplane of the fuel will depend on the occupancy of each pool. Accounting for the reduction in pool volume within the rack area, we estimate values of about 700 tonnes for Unit 1, 950 tonnes for Units 2-6, and 2,400 tonnes for the common pool.

The results clearly indicate that the Unit 4 pool was of greatest concern and that, unless there was a leak or other substantial loss of water due to the earthquake-induced sloshing, there was no immediate need to deal with

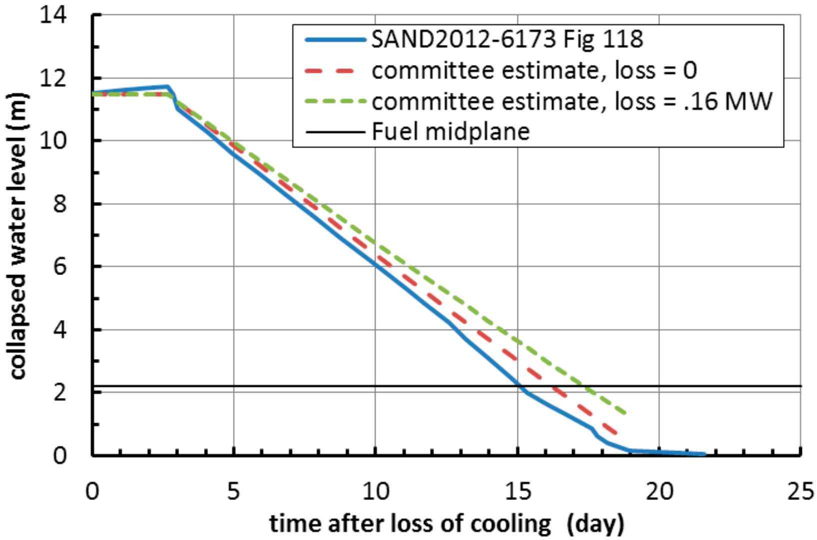

FIGURE 2B.4 Comparison of committee-estimated water levels in the Unit 4 pool with MELCOR computations (Gauntt et al., 2012, p. 192) of a full pool with no sloshing or water addition and spent fuel decay heat equivalent to that of the Unit 4 pool. Note: The committee estimates have been carried out for both zero energy loss by heat transfer and the value adopted by TEPCO.

the other pools for the first month after March 11, 2011. Unfortunately, the status of the pools was initially unknown so that, while these estimates are straightforward, it was difficult for the Fukushima Daiichi plant operators and TEPCO to have confidence in these results in the absence of any measurements of pool level and temperature.

To evaluate alternate outcomes of the Fukushima accident for Unit 4, one must evaluate the thermal power in the pool for times earlier than March 11. This was done using the Wigner-Way model of decay heat (Lewis, 2008, Equation 1.27)

$$\dot{Q}_{sf} = 0.0622\ \dot{Q}_{th}[t^{-0.2} - (t + t_{ic})^{-0.2}]$$

and adjusting the parameters $\dot{Q}_{th}$ = 2,390 MW and t_{ic} = 1,000 days[5] by fitting the decay heat to the TEPCO estimates (TEPCO, 2012a), which are

[5] The parameters in the equation are $\dot{Q}_{th}$, the reactor thermal power, and t_{ic}, the time the spent fuel has been at power in the core. Because there was a mix of different age fuel in the pool, the values obtained from the fitting process cannot be interpreted literally.

based on a realistic representation of reactor core neutron distribution and a detailed model of radionuclide generation and decay. The results of the fit are shown in Figure 2B.5.

Another application of this model is to determine the addition rate of water required to maintain steady-state conditions within the pool, i.e., constant temperature and mass. Applying these conditions to the mass and energy balance we obtain the required mass addition rate to maintain pool cooling:

$$\dot{M}_{add} = \frac{\dfrac{\dot{Q}'}{\Delta H_{fg}} + \dot{M}_{leak}}{1 + \dfrac{C_p \Delta T}{\Delta H_{fg}}}$$

As expected, the mass added must be sufficient to compensate for both the leak as well as the evaporation. The term in the denominator accounts for the thermal power used to heat the cooler added water to the pool temperature; evaluating this for a pool temperature of 90°C, we find that the

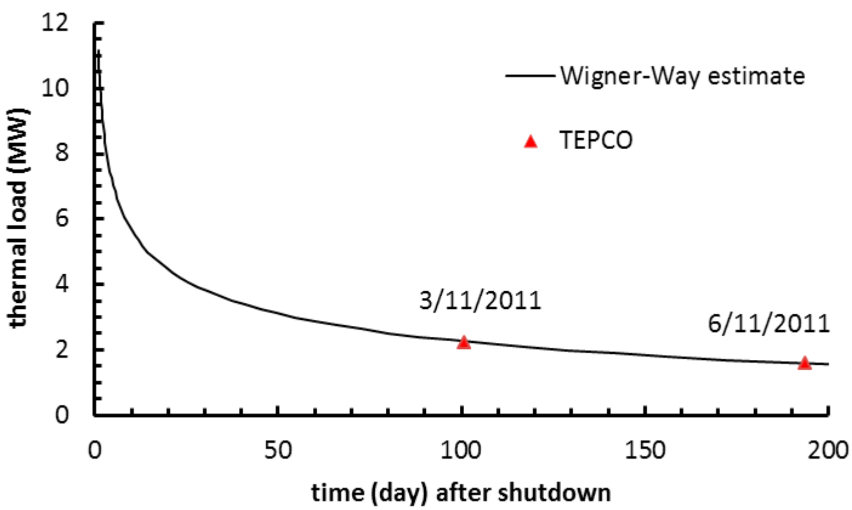

FIGURE 2B.5 Wigner-Way model of decay heat for the Unit 4 reactor core after reactor shutdown at the end of November 2010. Note: The data points are estimates by TEPCO based on the actual fuel inventory and a detailed model of the radionuclide decay.

denominator is less than or equal to 1.1 and makes a negligible correction to the intuitive result,

$$\dot{M}_{add} \approx \frac{\dot{Q}'}{\Delta H_{fg}} + \dot{M}_{leak} = \dot{M}_{vap} + \dot{M}_{leak}.$$

The first term on the right-hand side is just the rate of evaporation needed to keep the pool at constant temperature. For all but the smallest of leaks, the mass addition rate required to maintain the pool level constant will be determined by the size of the leak rather than by the evaporation rate needed for thermal equilibrium. This is because the rate of evaporation needed even for 10-day-old fuel is only 40 gpm (220 tonne/day).

APPENDIX 2C
Orifice Flow-Rate Correlation

The committee estimated leakage rates around the gate seals in the Unit 4 spent fuel pool using the conventional orifice flow rate correlation:

$$\dot{V} = C_f A_0 \sqrt{\frac{2\Delta P}{\rho}},$$

where $\dot{V}$ is the volumetric flow rate, C_f is the pressure drop coefficient (an empirical value, taken to be 0.6 for the purposes of this estimate), A_0 is the flow-passage area, ΔP is the pressure drop across the gates,[1] and ρ is the fluid density (in this case, the density of water in the pool).

The value for ΔP was estimated using the average of the hydrostatic pressure due to the water-level difference between the reactor well (h_1) and spent fuel pool (h_2) measured from the bottom of the spent fuel pool:

$$\Delta P = \frac{1}{2}(h_1 - h_2)\rho g,$$

where g is the acceleration due to gravity. Because of the gate design (see Figure 2C.1), leakage can occur when the water level in the reactor well is higher than in the spent fuel pool, that is, when ΔP is positive. The force of water created by a sufficiently higher level on the pool side than the well side pushes the gate toward the well side and squeezes an elastomer seal to stop the flow of water out of the pool.[2] When the water level on the well side is higher than in the pool, the gate is mounted such that the force due to the difference in water level can displace the gate sufficiently that the seal is not effective and water will flow from the well into the pool; see the discussion in TEPCO (2012a, Attachment 9-1, p. 3/9).

Computations were carried out with various dimensions for flow-passage area (A_0); the results for pool level history are insensitive to its exact value, resulting in variations in level within a band of ±0.5 m for effective seal openings between 0.1 and 5 mm (see Figure 2C.2). For this

[1] In applying the formula to cases where the flow is bidirectional, the absolute value of the pressure difference is used and the sign of the volume flow rate is taken to be the same as the sign of the pressure difference.

[2] When the level in the reactor well is slightly lower than that of the pool, TEPCO has proposed that sufficiently strong motion could dislodge the gate and allow water to flow from the pool to the well side; see the discussion of the effect of the January 1, 2012, earthquake in http://www.tepco.co.jp/en/nu/fukushima-np/roadmap/images/m120123_02-e.pdf.

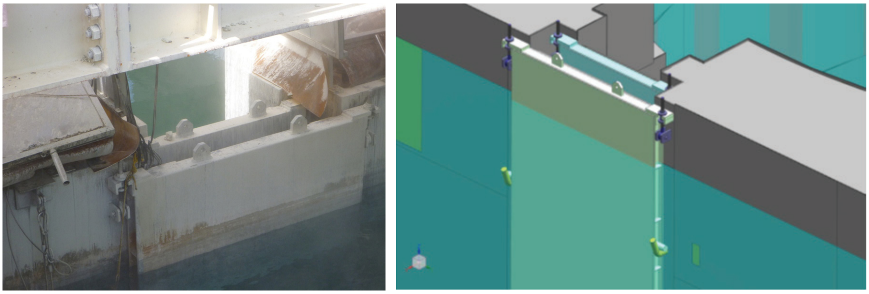

FIGURE 2C.1 *Left image:* View from the pool side of the two gates separating the Unit 4 spent fuel pool from the reactor well and dryer-separator pit. The sealing surface is located on the far side of each gate. *Right image:* Schematic illustration showing gate arrangements and attachments. Note: The gates are held in place by hooks on the wall separating the pool from the reactor well and are bolted at their tops to prevent floating. SOURCE: Photo taken by TEPCO on June 29, 2011. Available at http://photo.tepco.co.jp/en/date/2011/201106-e/110630-04e.html/. Image from TEPCO (2012c).

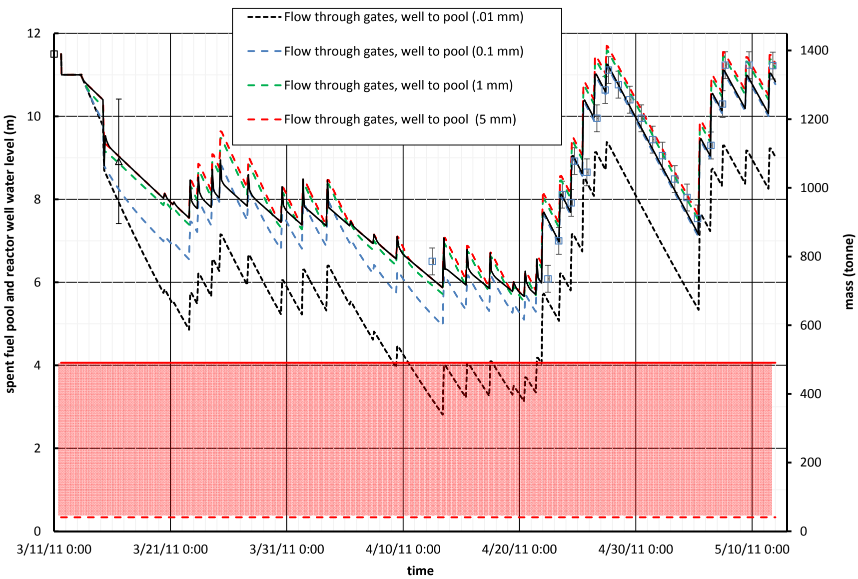

FIGURE 2C.2 Committee model for Unit 4 pool water level showing the effect of assuming four sizes of the gate passage opening on pool water-level history as well as assuming that prior to April 22 for one case (5 mm), the water could flow in both directions. Note: The results show that as long as the passageway was sufficiently large (> 1 mm), the results are not too sensitive to the choice of passageway size or limitations on directionality of flow prior to April 22.

range in seal openings, the total amount of water transferred between the reactor well and pool varied from about 650 to 710 tonnes, consistent with the measured reactor well water level on April 27, 2011. The water-level history before April 22 (when the gate seals appeared to become effective again[3]) was also insensitive to the assumptions about the directionality of water flow. Assuming that water could flow in both directions resulted in only a 0.5-m difference in pool level on April 22, 2011. In Figures 2.15 and 2.16, a value of 1.0 mm for the effective width of the flow-passage area was used and the height of the flow passage was taken to be equal to the difference in levels between the reactor well and spent fuel pool.

Once the leakage rate is computed, water levels in the reactor well and dryer-separator pit can be estimated using the mass balance as discussed in Appendix 2B. The reactor well and dryer-separator pit are estimated to have a combined volume of 1,400 m^3 and surface area of 172 m^2; these values are given by TEPCO (2012a, Attachment 9-1, Table 3; 2012b) and are consistent with other published data.[4] The committee-estimated water levels in the reactor well and dryer-separator pit are shown by the orange curve in Figure 2.15.[5]

[3] It is unclear why the gate seals regained effectiveness after April 22. The gates are just hung in place; their sealing depends on the force created by differential water levels between the pool and reactor well. This force increased substantially with the water addition to the pool on April 22.

[4] The values we have used are different than those given by other investigators such as EPRI (2012a), which reflects the difficulty of obtaining reliable data for complex geometries in the outage condition as well as uncertainty about the precise dimensions of the reactor buildings.

[5] The committee-computed water levels in the reactor well and dryer-separator pit are in good agreement with the TEPCO water-level measurement on April 27, 2011 (black square in Figure 2.15).

3

Fukushima Daiichi Nuclear Accident: Lessons Learned for Nuclear Plant Security

The primary focus of this chapter is on the security components of Study Charges 3 and 4 of the statement of task for this study (Sidebar 1.2 in Chapter 1):

- Lessons that can be learned from the accident to improve commercial nuclear plant safety and security systems and operations and
- Lessons that can be learned from the accident to improve commercial nuclear plant safety and security regulations, including processes for identifying and applying design-basis events for accidents and terrorist attacks to existing nuclear plants.

The safety portions of these tasks were addressed in this committee's phase 1 report (NRC, 2014).

The chapter also addresses the final part of Study Charge 4 on lessons learned on processes for identifying and applying design-basis events for accidents and terrorist attacks to existing nuclear plants.

3.1 LESSONS LEARNED FOR SECURITY AT U.S. NUCLEAR PLANTS

The March 11, 2011, Great East Japan Earthquake and tsunami caused extensive and long-lasting (days to weeks) damage to safety and security infrastructure at the Fukushima Daiichi plant. Written reports from the plant's operator, Tokyo Electric Power Company (TEPCO), describe the severe damage that occurred to plant safety systems following the earthquake and tsunami (TEPCO, 2011, 2012a; see also Investigation Committee, 2011, 2012). Detailed discussions of the accident and its impacts on the reactors and spent fuel pools at the Fukushima Daiichi plant are provided in the present committee's phase 1 report (NRC, 2014) and in Chapter 2 of the present report.

To the committee's knowledge, TEPCO has not publicly disclosed the impacts of the earthquake and tsunami on plant security systems. Nevertheless, the committee infers from TEPCO's written reports, as well as its own observations during a November 2012 tour of the Fukushima Daiichi plant, that security systems at the plant were substantially degraded by the earthquake and tsunami and the subsequent accident. There are three principal lines of evidence that support this inference:

- Physical damage. The areas surrounding Units 1-4 (see Sidebar 1.1 in Chapter 1) at the plant were flooded to depths of about 5.5 m by the tsunami; there was extensive damage to physical infrastructure in and surrounding the units, including damage to plant access controls in the owner-controlled and protected areas (see Chapters 3 and 4 in NRC, 2014).
- Electrical power. All offsite electrical power to the plant was lost following the earthquake, and DC power was eventually lost in Units 1-4 following the tsunami. Offsite AC power was not restored until 9 to 11 days later (see NRC, 2014, Table 4.1). Security equipment requiring electrical power was probably not operating continuously during this blackout period.
- Personnel. Plant workers, including workers monitoring the protected area of the plant, evacuated to higher ground just before the tsunami arrived on March 11, 2011 (TEPCO, 2012a, p. 163). Additionally, early on the fourth day of the accident (March 15), about 650 workers were temporarily evacuated from the plant. Among the evacuated workers was the Security Guidance Team, which was responsible for controlling plant access. This team did not return to the plant until the afternoon of that same day (TEPCO, 2012a, p. 166).[1] These evacuation events are described in the present committee's phase 1 report (NRC, 2014, pp. 107-108).

Tsunami damage and power losses likely affected the integrity and operation of numerous security systems, including lighting, physical barriers and other access controls, intrusion detection and assessment equipment, and communications equipment.

[1] Among the 70 workers who remained at the plant after the evacuation were workers responsible for plant monitoring (TEPCO, 2012a, p. 166).

The committee's observations about the impacts of the earthquake and tsunami on security at the Fukushima Daiichi plant led to one finding and recommendation, presented in the next section.

3.1.1 Finding and Recommendation

FINDING 3.1: Extreme external events and severe accidents such as occurred at the Fukushima Daiichi plant can cause widespread and long-lasting disruptions to security infrastructure, systems, and staffing at nuclear plants. Such disruptions can create opportunities for malevolent acts and increase the susceptibility of critical plant systems to such acts.

RECOMMENDATION 3.1: Nuclear plant operators and their regulators should upgrade and/or protect nuclear plant security infrastructure and systems and train security personnel to cope with extreme external events and severe accidents. Such upgrades should include

- **Independent, redundant, and protected power sources dedicated to plant security systems that will continue to function independently if safety systems are damaged;**
- **Diverse and flexible approaches for coping with and reconstituting plant security infrastructure, systems, and staffing during and following extreme external events and severe accidents; and**
- **Training of security personnel on the use of these approaches.**

The U.S. nuclear industry should consider expanding its Diverse and Flexible Coping Strategies (FLEX) capability to address this recommendation. The U.S. Nuclear Regulatory Commission (USNRC) should support industry's efforts by providing guidance on approaches and by overseeing independent review by technical peers (i.e., peer review).

3.1.2 Discussion

To the committee's knowledge, no significant security incidents occurred at the Fukushima Daiichi plant during or after the accident. Nevertheless, the events at the plant suggest an important lesson from the accident: *Extreme external events and severe accidents can have severe and long-lasting impacts on the security systems at nuclear plants. Such long-lasting disruptions can create opportunities for malevolent acts and increase the susceptibility of critical plant systems to such acts.*

Power and certain safety and security systems were unavailable at the Fukushima Daiichi plant for weeks after the March 11, 2011, earthquake and tsunami. Similar situations could occur as a result of other natural disasters. For example, a hurricane or destructive thunderstorm that spawned tornados could damage onsite and offsite power substations and high-voltage pylons, causing a loss of a nuclear plant's offsite power. The storm could also damage security fences, cameras, and other intrusion-detection equipment. Relief security officers and other site personnel may not be able to report to duty on schedule if storm-related damage was widespread in surrounding communities. An adversary could use this disruption to advantage in carrying out a malevolent act.

An extreme external event or severe accident at a U.S. nuclear plant could require the temporary suspension of security measures. USNRC regulations allow for such suspensions under the conditions specified in 10 Code of Federal Regulations (CFR) 73.55(p) (*Suspension of Security Measures*):

> (1) The licensee may suspend implementation of affected requirements of this section under the following conditions:
>
> (i) In accordance with §§ 50.54(x) and 50.54(y) of this chapter, the licensee may suspend any security measures under this section in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. This suspension of security measures must be approved as a minimum by a licensed senior operator before taking this action.
>
> (ii) During severe weather when the suspension of affected security measures is immediately needed to protect the personal health and safety of security force personnel and no other immediately apparent action consistent with the license conditions and technical specifications can provide adequate or equivalent protection. This suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager, before taking this action.
>
> (2) Suspended security measures must be reinstated as soon as conditions permit.
>
> (3) The suspension of security measures must be reported and documented in accordance with the provisions of § 73.71.

The regulations are specific about the conditions under which security at a nuclear plant can be suspended: to protect public health and safety (point (i) above) and protect the health and safety of security personnel (point (ii) above).

The regulations require that the suspended security measures be reinstated as soon as conditions permit. The Fukushima Daiichi accident illustrates that full restoration of security measures could potentially take days to weeks after an extreme external event or severe accident: Damaged security equipment must be restored and destroyed equipment must be replaced. During this interim period, security could be provided by increasing the size of the guard force at the plant to perform needed surveillance and access control functions if habitable conditions exist.

U.S. nuclear plants are required to have both onsite and offsite emergency response plans. Security personnel are key participants in the onsite emergency plan. 10 CFR 50.47 (*Emergency Plans*) requires that adequate staffing be maintained at all times to provide initial facility accident response in key functional areas. Interim Compensatory Measures Order EA-02-026,[2] issued after September 11, 2011, requires that sufficient personnel be available on each shift to implement security and emergency plans.

The assembly and accountability requirements during an emergency are normally implemented by members of the security force and utilize an accountability system based in the security computer that maintains normal logs of personnel entering and exiting the facility. The protective action options of sheltering and evacuation are combined with a consideration of the necessity for keeping specific technical or management personnel onsite. The security force assists in implementing site evacuations.

The committee's recommendation calls for upgrading and/or hardening nuclear plant security infrastructure, systems, and training to cope with extreme external events and severe accidents. The committee judges that the following three actions are needed:

1. Ensuring that there is adequate separation of plant safety and security systems so that the security systems can continue to function independently if safety systems are damaged. In particular, security systems need to have independent, redundant, and protected power sources so that they continue to function when normal plant power is unavailable.
2. Implementing diverse and flexible approaches for coping with and reconstituting plant security infrastructure, systems, and staffing during and following external events and severe accidents.
3. Training of security personnel on implementing approaches for reconstituting security infrastructure and systems.

With respect to point 1, the regulations in 10 CFR Part 73 (*Physical Protection of Plants and Materials*) require that intrusion detection and assessment equipment at the perimeter of the plant's protected area remain operable from an uninterruptible power supply in the event of the loss of

[2] This order is designated as Safeguards Information and is not available to the public.

normal power (10 CFR 73.55(i)(3)(vii)); similarly, the regulations require that nonportable communications equipment must remain operable from independent power sources in the event of the loss of normal power (10 CFR 73.55(j)(5)). However, the regulations do not specify the performance requirements for these backup power supplies.[3] These backup supplies need to be adequately protected and sized to cope with a long-duration event such as occurred at the Fukushima Daiichi plant.

With respect to points 2 and 3, the U.S. nuclear industry has developed and is currently implementing its FLEX initiative (NEI, 2012) to augment the coping capabilities at nuclear plants to external beyond-design-basis events. The strategy has four elements:

1. To have portable backup equipment capable of providing water and power to the reactor. Such equipment includes, for example, electrical generators, batteries, and battery chargers; compressors; pumps, hoses, and couplings; equipment for clearing debris; and equipment for temporary protection against flooding.
2. To stage this equipment in locations both on- and offsite where it will be safe and deployable.
3. To develop procedures and guidance for implementing FLEX.
4. To develop programmatic controls that will ensure personnel are well trained and equipment is maintained.

Each plant is responsible for developing implementation procedures for the protection and deployment of equipment, procedural interfaces, and utilization of offsite resources.

The committee sees an opportunity for industry to expand its FLEX initiative to include critical security-related equipment, such as access control, intrusion detection and assessment, communications, and portable-lighting equipment. This equipment would need to be sufficiently standardized so that it could be used across the U.S. nuclear plant fleet and adequately protected against extreme external events, severe accidents, and sabotage.[4] Security personnel at U.S. plants would need to be trained on the use of this equipment if it were different from existing equipment at their plants.

[3] Regulatory Guide 5.44 (USNRC, 1997a, p. 5.44-6) states that "Emergency power should be capable of sustaining operation without external support for . . . a site-specific period of time determined according to station blackout criteria for power reactor facilities." Additionally, USNRC (2011a) states that "[t]he capability of the emergency/backup power source to sustain security system operations should be based on the timeframe to restore primary power as derived through a site specific analysis" (p. 9.2).

[4] This applies to equipment located at nuclear plants as well as equipment located at regional FLEX facilities.

3.2 LESSONS LEARNED FOR IDENTIFYING AND APPLYING DESIGN-BASIS EVENTS FOR ACCIDENTS AND TERRORIST ATTACKS TO EXISTING NUCLEAR PLANTS

The Committee's phase 1 report described a *design-basis event* as

> "a postulated event that a nuclear plant system, including its structures and components, must be designed and constructed to withstand without a loss of functions necessary to protect public health and safety. An event that is 'beyond design basis' has characteristics that could challenge the design of plant structures and components and lead to a loss of critical safety functions." (NRC, 2014, p. 9)

The USNRC uses the design-basis concept for regulating both the safety and security of commercial nuclear plants:

- The USNRC uses the *design-basis accident* (DBA) concept in its safety-related regulations. DBAs describe a specified set of failures or abnormal events, for example equipment malfunctions, which must be considered in the design of a nuclear plant. Plant safety systems must be designed to allow plant operators to recover the plant to a safe state following such malfunctions. The committee's phase 1 report (NRC, 2014) discusses the application of design-basis events for accidents to existing nuclear plants (see especially Sidebar 1.2 in Chapter 1 and Section 5.2 in Chapter 5).
- The USNRC uses the *design-basis threat* (DBT) concept in its security-related regulations. DBTs describe a specified set of adversary attributes that must be considered in the design of plant security systems. The USNRC has established DBTs for radiological sabotage[5] and for theft or diversion of formula quantities of strategic special nuclear materials.[6] These requirements are defined in 10 CFR Part 73 (*Physical Protection of Plants and Materials*).

The USNRC assesses licensees' compliance with the regulations through a number of means. Some of these are discussed in Chapter 4 of the present report (see especially Section 4.1.3).

[5] Radiological sabotage is defined in 10 CFR 73.2 as "any deliberate act directed against a plant or transport . . . or against a component of such a plant or transport which could directly or indirectly endanger the public health and safety by exposure to radiation."

[6] Special nuclear material includes plutonium, uranium-233, or uranium enriched in the isotopes uranium-233 or uranium-235. Formula quantity is defined as 5,000 grams or more, in any combination, of grams U-235 + 2.5 * grams U-233 + grams plutonium.

Generic characteristics of the DBT for radiological sabotage are described in 10 CFR 73.1(a)(1):

> Radiological sabotage. (i) A determined violent external assault, attack by stealth, or deceptive actions, including diversionary actions, by an adversary force capable of operating in each of the following modes: A single group attacking through one entry point, multiple groups attacking through multiple entry points, a combination of one or more groups and one or more individuals attacking through multiple entry points, or individuals attacking through separate entry points, with the following attributes, assistance and equipment:
>
> (A) Well-trained (including military training and skills) and dedicated individuals, willing to kill or be killed, with sufficient knowledge to identify specific equipment or locations necessary for a successful attack;
>
> (B) Active (e.g., facilitate entrance and exit, disable alarms and communications, participate in violent attack) or passive (e.g., provide information), or both, knowledgeable inside assistance;
>
> (C) Suitable weapons, including handheld automatic weapons, equipped with silencers and having effective long range accuracy;
>
> (D) Hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or for otherwise destroying reactor, facility, transporter, or container integrity or features of the safeguards system; and
>
> (E) Land and water vehicles, which could be used for transporting personnel and their hand-carried equipment to the proximity of vital areas; and
>
> (ii) An internal threat; and
>
> (iii) A land vehicle bomb assault, which may be coordinated with an external assault; and
>
> (iv) A waterborne vehicle bomb assault, which may be coordinated with an external assault; and
>
> (v) A cyber attack.

The detailed characteristics of the DBT—for example the number of attackers, their training, and weaponry—are determined by USNRC commissioners based on USNRC staff analyses of terrorist motivations, capabilities, and technical means. The information used in these analyses is obtained from U.S. law enforcement, homeland security, and intelligence agencies.

The DBT is not designed to be the worst-case threat. It simply defines the upper bound within the total threat environment against which a nuclear plant licensee is required to protect. The responsibility for protecting against beyond-DBT threats is the responsibility of federal, state, and local agencies. The National Infrastructure Protection Plan (NIPP; DHS, 2013) describes how governmental and private-sector participants in the critical infrastructure community (including the nuclear power industry) work together to manage security risks.

The NIPP includes 16 Sector-Specific Plans (SSPs), including a Nuclear SSP (DHS, 2010). The Nuclear SSP covers the following Critical Infrastructure and Key Resources (CI/KR): nuclear power reactors and research and test reactors; fuel fabrication plants; civilian nuclear materials use; and transportation, storage, and disposal of nuclear material and waste. The 2010 version of the plan acknowledges that "some threats are beyond what is reasonable to expect CI/KR owners and operators to protect against by themselves."[7]

The U.S. Department of Homeland Security (DHS) has the responsibility for nuclear CI/KR protection in cooperation with the USNRC. Government and sector coordinating councils have been established to share information and coordinate security strategies, activities, policies, and communications. The Government Coordinating Council comprises representatives from DHS, USNRC, the Federal Bureau of Investigation, and the Department of Energy. The private coordinating council consists of representatives from the nuclear industry.

3.2.1 Discussion

The committee obtained written information about the NIPP and Nuclear SSP but did not have enough time to obtain in-depth briefings on operational details and responsibilities. The committee also did not have adequate time to carry out an in-depth analysis of processes for identifying and applying design-basis events for accidents and terrorist attacks to existing nuclear plants. Consequently, the committee provides observations about these processes rather than formal findings and recommendations.

The committee's first observation concerns the application of the design-basis concept to nuclear plants: DBAs and DBTs are not intended to cover all safety and security events that can arise at a nuclear plant; rather, they are intended to guide the development of plant safety and security systems. Beyond-DBAs are managed in a number of different ways, for example, through the layering of safety and security capabilities (defense-in-depth; see Appendix 3A), or through operator training (B.5.b and Severe Accident

[7] The Nuclear SSP was being updated when the present report was being finalized.

Management Guidelines) and pre-positioned equipment (FLEX).[8] Beyond-DBT security threats are managed by the plant's security forces with assistance from local law enforcement and possibly from other government agencies through the NIPP.

The nuclear industry conducts *safety* risk assessments on a routine basis to identify potential beyond-DBA scenarios and manage their consequences. However, there is no equivalent process in place for conducting *security* risk assessments to identify beyond-DBTs and manage their consequences. The committee provides further analysis and a recommendation of security risk assessment in Chapters 4 and 5.

The committee's second observation concerns the applicability of the DBT concept to protecting nuclear plants against asymmetric threats.[9] An adversary who lacks the strength, weaponry, and training of the nuclear plant's security forces might utilize attack strategies that do not require direct confrontations with those forces. For example, an adversary might choose to attack perceived weak points in the plant's support infrastructure (e.g., offsite power and water supplies, key personnel) rather than mounting a direct assault on the plant. The goals of such asymmetric attacks might be to cause operational disruptions, economic damage, and/or public panic rather than radiological releases from a plant's reactors or spent fuel pools. In fact, such attacks would not necessarily need to result in any radiological releases to be considered successful.

Offsite power substations, piping, fiber optic connection points, and other essential systems provide an adversary the opportunity to inflict damage with very little personal risk and without confronting a nuclear plant's security forces.[10] The psychological effects of such attacks, even if these do not result in the release of radioactive material, might have consequences comparable to or greater than the actual physical damage. In the extreme, such attacks could lead to temporary shutdowns of, or operating restrictions on, other nuclear plants until security enhancements could be implemented. (Japan shut down all its nuclear power reactors and briefly entertained the dismantlement of its nuclear power industry due to public pressure following the Fukushima Daiichi accident.)

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[8] See Chapter 5 and Appendixes F and H in NRC (2014) for information about these capabilities.

[9] The term *asymmetry* refers to dissimilarities in the capabilities, strategies, and/or tactics between an adversary and a defending force, for example, a terrorist cell intent on attacking a nuclear plant and that plant's security forces.

[10] Some rehearsals of this type of attack may have already taken place. In April 2013, one or more persons attacked a power transformer yard near San Jose, California, with high-powered rifles (Parfomak, 2014). The suspects are still at large. It is not clear whether their attack was simple vandalism or a rehearsal for a possible future attack on the U.S. power grid.

The DBT is not explicitly designed to address asymmetric threats. Rather, these threats are intended to be addressed by a plant's industrial security[11] programs.

Detailed information about the evolution of the accident at the Fukushima Daiichi plant and its compromised safety systems is widely available on the Internet and in reports such as this one. This information could be used by terrorists to plan and carry out asymmetric attacks on nuclear plants in hopes of creating similar cascading failures. The security risk assessment or CARVER (Criticality + Accessibility + Recuperability + Vulnerability + Effect + Recognizability) analysis described in Chapter 4 could identify asymmetric scenarios of potential concern and suggest ways to manage them.

[11] Industrial security is used to protect industrial facilities and equipment against unauthorized access, sabotage, espionage, and malicious manipulation. Industrial security at U.S. nuclear plants is implemented by licensees to protect their interests against security events that could result in operational disruptions but would not result in radiological releases.

APPENDIX 3A
Security Systems at Nuclear Plants

The impact of the earthquake and tsunami on the Fukushima Daiichi Nuclear Power Plant's security systems reinforces the need to ensure that facility security systems are (1) effective, (2) robust and resilient, (3) redundant and overlapping, and (4) readily recoverable.

3A.1 EFFECTIVENESS

To be effective, security systems must be designed and implemented to meet the "Five D's" of security: Deter, Detect and Assess, Delay, Deny, and Defend/Defeat a threat. Together these attributes define a "defense-in-depth" approach to security.

At the outermost boundary of a facility, a perimeter fence defines the owner-controlled area. It serves to *deter* persons, both via notice not to enter (through signage) and as an initial physical obstacle to entry. A person entering the facility by passing through or over the fencing is assumed to have intent to enter without authorization.

Various sensor systems can then be employed at (or just before or beyond) the perimeter fence to *detect* an intruder and to *assess* an intruder's intent and capabilities (e.g., whether the intruder is carrying a weapon). Additional barriers may be emplaced further inside the property to *delay* the intruder and to the extent possible *deny* further access, allowing time for the security force to respond.

Finally, an effective security system includes a well-trained and well-armed response force that may be deployed to *defend* against and *defeat* the threat before any sabotage occurs. The security system must also provide for alarm or notification to offsite forces to assist in addressing the threat and to contain any intruders attempting to leave the area.

3A.2 ROBUSTNESS AND RESILIENCE

Physical protection systems must be hardened to withstand extreme natural and accidental events, as well as physical attack. Cameras, sensors, and other systems must be powered by an uninterruptable power source, independent of the power sources used for routine and emergency power for plant safety systems. Ideally each subsystem within the overall security system must have its own independent power supply to prevent the loss of all systems concurrently.

3A.3 REDUNDANT AND OVERLAPPING

Security systems serve overlapping and redundant functions. An alarm by one sensor system must immediately be assessed by a second system. For example, detectors at the perimeter fence, such as vibration, e-field, and microwave, must be assessed using security cameras or other systems to confirm the attempted (or successful) intrusion.

3A.4 READILY RECOVERABLE

In the event of a catastrophic event or attack, security systems must be designed and installed to be quickly reconstituted. Hardened power and fiber optic cables must permit "plug-and-play" installation of replacements for inoperable equipment. Reestablishment of security is critical because an adversary who might otherwise be deterred from attacking a site might be encouraged to carry out an attack at a compromised facility.

4

Reevaluation of Findings and Recommendations from Previous NAS Reports

The focus of this chapter is on the second charge of the study task (Sidebar 1.2 in Chapter 1), which calls for a

> "Reevaluation of the conclusions from previous NAS studies on safety and security of spent nuclear fuel and high-level radioactive waste storage, particularly with respect to the safety and security of current storage arrangements and alternative arrangements in which the amount of commercial spent fuel stored in pools is reduced."

The "previous NAS studies" referred to in this task is a single study carried out in 2003-2004 at the request of the U.S. Congress. That study addressed the four tasks shown in Sidebar 4.1 and produced two reports:

- A report containing classified and other security-related information,[1] hereafter referred to as the *classified report* (NRC, 2004), and
- An abbreviated version of this classified report that was suitable for unrestricted public release, hereafter referred to as the *public report* (NRC, 2006[2]).

[1] That is, Safeguards Information, which is protected from unauthorized disclosure under Section 147 of the Atomic Energy Act, as well as sensitive unclassified non-safeguards information, referred to by many federal agencies as "official use only" information, which is restricted from public release through the Freedom of Information Act.

[2] Completion and release of the public report was delayed because of an extended security review by the sponsoring agency (U.S. Nuclear Regulatory Commission).

SIDEBAR 4.1
Statement of Task for NRC (2006) Report

The issues to be addressed by this study are specified in the Energy and Water Development Conference Report and are as follows:

1. Potential safety and security risks of spent nuclear fuel presently stored in cooling pools at commercial reactor sites.
2. Safety and security advantages, if any, of dry cask storage versus wet pool storage at these reactor sites.
3. Potential safety and security advantages, if any, of dry cask storage using various single-, dual-, and multipurpose cask designs.
4. In light of the September 11, 2001, terrorist attacks, this study will explicitly consider the risks of terrorist attacks on these materials and the risk these materials might be used to construct a radiological dispersal device.

The public report is similar in content to the classified report and contains all of its findings and recommendations. However, redactions and wording modifications were made to the classified report, including its findings and recommendations, to remove classified and other security-related information.

The discussions in this chapter are referenced primarily to the public report (NRC, 2006). Where necessary for completeness, identification of classified and other security-related information in the classified report (NRC, 2004) is made by reference to specific sections and page numbers in that report.

Table 4.1[3] summarizes the committee's reevaluation of the findings and recommendations in the public report (NRC, 2006). The left-hand column of the table displays the findings and recommendations in the public report (NRC, 2006) organized by their order of presentation in that report. The committee's reevaluation is presented in the right-hand column of the table, also in the form of findings and recommendations. To avoid confusion, these findings and recommendations are numbered using a scheme different from those in the National Research Council public report (NRC, 2006).[4]

[3] The table appears on pp. 102-110 at the end of this chapter because of its length.

[4] The findings and recommendations in NRC (2006) are numbered using the format *xy*, where *x* corresponds to the task number in NRC (2006) (i.e., tasks 1-4) and *y* is a serial letter (i.e., A-E). In the present report, the findings and recommendations are numbered using the format *4.z*, where *z* is a serial number (1, 2, . . .).

In performing its reevaluation, the present committee paid particular attention to the disposition of recommendations in the public report (NRC, 2006) by the U.S. Nuclear Regulatory Commission (USNRC) and the nuclear industry. The committee gathered information from the USNRC, nuclear industry, and independent analysts (see Appendix B) to determine (1) whether and how these recommendations were addressed and (2) what additional actions, if any, are needed.

The following sections provide a discussion of the committee's reevaluations. The sections are organized identically to those in Table 4.1. Detailed supporting information for some of the present committee's findings and recommendations in this chapter is provided in Appendix A and Chapters 5-7.

4.1 TERRORIST ATTACKS ON SPENT FUEL STORAGE OR THEFT OF SPENT FUEL

Chapter 2 of NRC (2006) addresses the fourth study task in Sidebar 4.1:

> "Explicitly consider the risks of terrorist attacks on [spent fuel] and the risk these materials might be used to construct a radiological dispersal device."[5]

The chapter provides background information on risk assessment and a brief discussion of possible terrorist motivations for attacking a nuclear plant and its spent fuel storage facilities. The report concluded that the terrorist attack risks could not be addressed "using quantitative and comparative risk assessments." Instead, the report examined "a range of possible terrorist attack scenarios in terms of (1) their potential for damaging spent fuel pools and dry storage casks; and (2) their potential for radioactive material releases" (NRC, 2006, p. 27). The report provided three findings and three recommendations to address the fourth study task (see Table 4.1):

- **Finding 2A** (NRC, 2006) notes that the probability of terrorist attacks on spent fuel storage could not be assessed quantitatively or comparatively and that spent fuel storage facilities could not be dismissed as targets for such attacks.
- **Finding 2B** (NRC, 2006) notes that the likelihood that terrorists could steal enough spent fuel for use in a significant radiological dispersal device is small. **Recommendation 2B** (NRC, 2006) encouraged the USNRC to review and upgrade, where necessary, its requirements for protecting spent fuel rods not contained in fuel

[5] That is, a device that disperses radioactive material using explosives or other means.

assemblies from knowledgeable insiders,[6] especially in facilities where fuel rods or portions of rods are being stored in pools.

- **Finding 2C** (NRC, 2006) notes that a number of security improvements at nuclear plants have been instituted since the events of September 11, 2001, but that the USNRC did not provide enough information to evaluate the effectiveness of these procedures for protecting stored spent fuel. **Recommendation 2C** (NRC, 2006) encouraged the performance of an independent[7] assessment of surveillance and security measures for protecting stored spent fuel.

The committee's reevaluation of these findings and recommendations is provided in the following sections.

4.1.1 Reevaluation of Finding 2A from NRC (2006)

The present committee agrees with NRC (2006) that there are technical challenges associated with identifying terrorist attack scenarios and quantifying their likelihoods. However, the committee judges that the NRC (2006) report's focus on quantification challenges is too narrow a perspective for judging the usefulness of applying risk assessment methods to nuclear plant security. The present committee finds (see **Finding 4.1** in Table 4.1) that the understanding of security risks at nuclear power plants and spent fuel storage facilities can be improved through risk assessment. Assessments that focus on the risk triplet—scenarios, likelihoods, and consequences—can contribute useful security insights for improving the protection of facilities and operations. Chapter 5 provides the committee's detailed rationale for this finding.

The present committee received a briefing from USNRC staff on ongoing and planned future work by the agency and the nuclear industry on development and application of risk assessment to nuclear plant security. The committee was encouraged to learn that the agency is working on this issue. The committee also recognizes that support from USNRC management and from the nuclear industry will be essential to the success of this effort. To encourage further progress, the present committee recommends (**Recommendation 4.1A**) that the U.S. nuclear industry and the USNRC strengthen their capabilities for identifying, evaluating, and managing the risks from terrorist attacks. The committee also recommends (**Recommendation 4.1B**) that the USNRC sponsor a spent fuel storage security risk assessment of sufficient scope and depth to explore the benefits of this

[6] An *insider* is a person who is authorized to have physical and/or cyber access to nuclear plant facilities and systems and is working alone or with others to attack the plant.

[7] That is, independent of the USNRC and the nuclear industry.

methodology for enhancing security at U.S. nuclear plants. This assessment should be subjected to independent review by technical peers (i.e., peer review) as part of the development process. See Section 5.1.1.8 in Chapter 5 of the committee's phase 1 report (NRC, 2014) for a discussion of peer review.

4.1.2 Reevaluation of Finding and Recommendation 2B from NRC (2006)

The committee agrees with Finding 2B in NRC (2006) that the "likelihood terrorists could steal enough spent fuel for use in a radiological dispersal device is small" for the same reasons described in pp. 33-34 of that report. The committee finds (**Finding 4.2**) that the USNRC has made good progress in upgrading its requirements for protecting spent fuel rods not contained in assemblies: The USNRC has taken steps to improve inventory recordkeeping and controls, enhance inspections, and update regulatory guidance for control and accounting of spent fuel rods and rod fragments. Additionally, the USNRC is undertaking a rulemaking to clarify and strengthen material control and accounting requirements for these materials.

4.1.3 Reevaluation of Finding and Recommendation 2C from NRC (2006)

The committee finds (**Finding 4.3**) that the USNRC has not obtained the independent examination of surveillance and security measures for protecting stored spent fuel that was recommended by NRC (2006). USNRC staff told the committee that the agency obtains independent reviews of security readiness at nuclear plants through its force-on-force testing program, and also that the agency receives independent advice from the Advisory Committee on Reactor Safeguards[8] (ACRS). The committee agrees that the force-on-force testing program is important for assessing the training and operational readiness of a plant's security forces. However, this testing does not in itself constitute the independent assessment of a plant's surveillance and security measures recommended by NRC (2006). Moreover, the ACRS does not review USNRC security matters.

[8] The ACRS has oversight on all safety aspects of nuclear plants including spent fuel storage facilities. A recent example of such oversight is the July 18, 2013, ACRS letter report concerning the USNRC's Spent Fuel Study (see ACRS, 2013). Committee member Michael Corradini is a member of the ACRS. Additional information about the ACRS can be found at http://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html.

Commission staff provided the present committee with written information[9] and briefings on many of the agency's security requirements and programs, including

- Physical security at nuclear plants,
- Security performance characteristics and training,
- Personnel access authorization at nuclear plants,
- Design-basis threat for commercial power reactors,
- Mitigation of the insider threat at commercial power reactors, and
- Material control and accounting of spent fuel assemblies and fuel rods.

It was clear from these briefings that the USNRC has an extensive set of requirements and programs for ensuring the protection of nuclear plants and their spent fuel storage facilities. However, the present committee was unable to assess the effectiveness of these requirements and programs: Such an assessment was not within the scope of the present study and, moreover, it would require a dedicated effort with a committee having more focused physical, cyber, and personnel security expertise than exists on the present committee.

If the USNRC carries out the independent examination of surveillance and security measures that was recommended by NRC (2006), then the present committee recommends (**Recommendation 4.3**) that it include an examination of the effectiveness of measures for addressing the *insider threat*. Several programs have been put into place by the USNRC and the nuclear industry to address the trustworthiness and reliability of individuals with access to nuclear plants to minimize the potential for malevolent actions, including

- Fitness for duty,
- Access authorization, and
- Behavior observation.

The committee received written documents[10] and briefings on these measures from USNRC staff.

The USNRC requires licensees to implement an Insider Mitigation Program to oversee and monitor the initial and continuing trustworthiness and reliability of individuals having unescorted access in protected or vital

[9] The written documents contain information that is exempt from public release through the Freedom of Information Act. Consequently, their content cannot be described in this report.

[10] The written documents contain Safeguards Information and other security-related information, so their content cannot be described in this report.

areas of nuclear plants. There is a long-standing assumption by the USNRC that this program reduces the likelihood of an active insider (GAO, 2006). USNRC staff was not able to provide an explanation that was adequate to the committee on how it assesses the effectiveness of these measures for mitigating the insider threat. Moreover, to the committee's knowledge, there are no programs in place at the USNRC to specifically evaluate the effectiveness of these measures for mitigating the insider threat.

4.2 SAFETY AND SECURITY OF POOL STORAGE

Chapter 3 of NRC (2006) addresses the first task of that study (Sidebar 4.1), which calls for an assessment of the

> "Potential safety and security risks of spent nuclear fuel presently stored in cooling pools at commercial reactor sites."

The safe storage of spent fuel in pools depends critically on keeping the fuel covered with water. This fact was understood more than 40 years ago and was powerfully reinforced by the Fukushima Daiichi accident. If pool water is lost through an accident or terrorist attack (such occurrences are referred to as *loss-of-pool-coolant event*s), then the fuel can become uncovered, possibly leading to fuel damage, including *zirconium cladding fires*,[11] that can result in the release of radioactive materials to the environment. NRC (2006) reviewed work carried out by the USNRC and others to better understand how stored fuel can become uncovered as well as the consequences of such uncovery.

Chapter 3 of NRC (2006) provides background information on spent fuel pool storage, examines potential initiating mechanisms for loss-of-pool-coolant events, and examines the potential consequences of such events. The chapter contains five findings and three recommendations (see Table 4.1):

- **Finding 3A** (NRC, 2006) notes that pool storage is required at all operating commercial nuclear plants to cool newly discharged spent fuel.
- **Finding 3B** (NRC, 2006) notes that a terrorist attack that partially or completely drained a spent fuel pool could, under some conditions, lead to a propagating zirconium cladding fire and the release of large quantities of radioactive materials to the environment.

[11] The term *zirconium cladding fire* is used to describe the self-sustaining oxidation of zirconium fuel cladding. This oxidation results in a temperature runaway that can generate enough heat to melt the fuel pellets. See Sidebar 2.2 in Chapter 2.

- **Finding 3C** (NRC, 2006) notes that it appears to be feasible to reduce the likelihood of zirconium cladding fires following a loss-of-pool-coolant event using readily implemented measures.
- **Finding 3D** (NRC, 2006) notes that the vulnerabilities of spent fuel pools to terrorist attacks are plant-design specific and can be understood only by examining the characteristics of spent fuel storage at each plant.
- **Finding 3E** (NRC, 2006) notes that progress has been made by the USNRC and others to understand the potential vulnerabilities and consequences of terrorist attacks on spent fuel pools but that additional work is needed.
- **Recommendations 3E-1** (NRC, 2006) specifies the additional analyses that should be carried out by the USNRC to improve the understanding of vulnerabilities and consequences of terrorist attacks on pool storage.
- **Recommendation 3E-2** (NRC, 2006) specifies two measures that should be taken by the USNRC to reduce the consequences of loss-of-coolant events.

Finding 3A and Finding 3D of NRC (2006) are statements of fact that require no reevaluation. Consequently, they are not discussed further. The other findings and recommendations are reevaluated in the followings sections.

4.2.1 Reevaluation of Finding 3B from NRC (2006)

NRC (2006) considered four general types of terrorist attack scenarios:

- Air attacks using large civilian aircraft or smaller aircraft laden with explosives,
- Ground attacks by groups of well-armed and well-trained individuals,
- Attacks involving combined air and land assaults, and
- Thefts of spent fuel for use by terrorists (including knowledgeable insiders) in radiological dispersal devices.

The report noted that

> ". . . only attacks that involve the application of large energy impulses or that allow terrorists to gain interior access have any chance of releasing substantial quantities of radioactive material. This further restricts the scenarios that need to be considered. For example, attacks using rocket-propelled grenades (RPGs) of the type that have been carried out in Iraq against U.S. and coalition forces would not likely be successful if the intent

> of the attack is to cause substantial damage to the facility. Of course, such an attack would get the public's attention and might even have economic consequences for the attacked plant and possibly the entire commercial nuclear power industry." (NRC, 2006, p. 30)

The concluding sentence speaks to terrorist intent and metrics for success. That is, if the intent of a terrorist attack is to instill fear into the population and cause economic disruption, then an attack need not result in any release of radioactive material from the plant to be judged a success. The classified report (NRC, 2004) identified particular terrorist attack scenarios that were judged by its authoring committee to have the potential to damage spent fuel pools and result in the loss of water coolant (see Section 2.2 in NRC, 2004). The present committee asked USNRC staff whether any of these attack scenarios had been examined further since NRC (2004) was issued. Staff was unable to present the committee with any additional technical analyses of these scenarios. Consequently, the present committee finds (**Finding 4.4**) that the USNRC has not undertaken additional analyses of terrorist attack scenarios to provide a sufficient technical basis for a reevaluation of Finding 3B in NRC (2004).

The present committee did not have enough information to evaluate the particular terrorist attack scenarios identified in NRC (2004) and therefore cannot judge their potential for causing damage to spent fuel pools. The committee notes, however, that new remote-guided aircraft technologies have come into widespread use in the civilian and military sectors since NRC (2004) was issued. These technologies could potentially be employed in the attack scenarios described in NRC (2004).

Other types of threats, particularly insider and cyber threats, have grown in prominence since NRC (2004) was issued. There is a need to more fully explore these threats to understand their potential impacts on nuclear plants. The committee-recommended risk assessment (see Finding 4.1 and associated Recommendations 4.1A and 4.1B in Table 4.1) would be an appropriate way to explore these threats.

4.2.2 Reevaluation of Finding 3C from NRC (2006)

NRC (2006) identified three measures that appear to have particular merit for reducing the likelihood of zirconium cladding fires following loss-of-pool-coolant events:

1. Reconfiguring spent fuel in the pools (i.e., redistribution of high-decay-heat assemblies so that they are surrounded by low-decay-heat assemblies) to more evenly distribute decay-heat loads and enhance radiative heat transfer;

2. Limiting the frequency of offloads of full reactor cores into spent fuel pools, requiring longer shutdowns of the reactor before any fuel is offloaded, and providing enhanced security when such offloads must be made; and
3. Developing redundant and diverse response systems to mitigate loss-of-pool-coolant events that would be capable of operation even if the pool or overlying building were severely damaged.

The present committee received briefings and technical reports from USNRC and Sandia National Laboratories staff on additional technical analyses and physical experiments that have been carried out since NRC (2004) was released. Some of the key reports examined by the committee are described in Chapter 6. The committee finds (**Finding 4.5**) that these analyses confirm that reconfiguring spent fuel in pools to more evenly distribute heat loads and enhance heat transfer can be an effective strategy for reducing the likelihood of fuel damage and zirconium cladding fires following loss-of-pool-coolant events. If a loss-of-pool-coolant event results in fuel uncovery, then reconfiguration may provide additional time for mitigating actions to be taken. However, reconfiguring spent fuel in pools does not completely eliminate the risks of zirconium cladding fires, particularly during certain periods following reactor shutdowns and for certain fuel and water configurations in the pool.[12] Additional discussion of these issues is provided in Chapter 6.

4.2.3 Reevaluation of Finding 3E and Recommendations 3E-1 and 3E-2 from NRC (2006)

The USNRC and its technical contractor, Sandia National Laboratories, have performed physical experiments and computer analyses (the latter using the Methods for Estimation of Leakages and Consequences of Releases [MELCOR] code; see Sidebar 6.1 in Chapter 6) to elucidate the phenomenology and consequences of loss-of-coolant events in spent fuel pools. These experiments and analyses focused on determining whether runaway oxidation of the fuel cladding (i.e., zirconium cladding fires) could develop in the stored fuel assemblies and propagate to other assemblies in the pool; whether specific configurations of fuel in the pool could delay or prevent this oxidation reaction from occurring; and whether certain mitigating strategies are effective for preventing this reaction from occurring. A description of these studies and some key results and remaining questions are provided in Chapter 6.

[12] Specific shutdown times and fuel and pool water configurations are considered by the USNRC to be security-related information and therefore are not disclosed in this report.

The present committee finds (**Finding 4.6**) that these additional experiments and analysis have substantially improved the state of knowledge concerning spent fuel behavior following partial or complete loss of pool water. The committee recommends (**Recommendation 4.6**) that the USNRC sponsor an end-to-end validation[13] of the MELCOR code for modeling loss of coolant in spent fuel pools and validate key submodels. The committee also finds (**Finding 4.7**) that the USNRC has not analyzed the potential vulnerabilities of spent fuel pools to the specific terrorist attack scenarios identified in NRC (2004) (see Section 4.2.1 in this chapter).

The committee finds (**Finding 4.8**) that the USNRC and the nuclear industry have made good progress in implementing the actions in Recommendation 3E-1 in NRC (2006). The committee recommends (**Recommendation 4.8**) that the USNRC and industry take additional steps to further reduce risks of zirconium cladding fires and improve mitigation capabilities. Additional discussion of these issues is provided in Chapter 6.

4.3 SAFETY AND SECURITY OF DRY CASK STORAGE AND COMPARISON WITH POOL STORAGE

Chapter 4 of NRC (2006) addresses the first study task in that report (Sidebar 4.1), which calls for an assessment of the

> "Safety and security advantages, if any, of dry cask storage versus wet pool storage at [commercial] reactor sites."

The chapter provides background information on dry cask storage, its potential risks, as well as potential advantages over pool storage. The chapter contains five findings and one recommendation (see Table 4.1):

- **Finding 4A** (NRC, 2006) notes that although there are differences in the robustness of different dry cask designs, the differences are not large.
- **Finding 4B** (NRC, 2006) notes that additional steps can be taken to make dry casks less vulnerable to terrorist attacks. **Recommen-**

[13] That is, the validation of key parameters in all phases of an analyzed scenario. For example, for a scenario involving indefinite loss of cooling in a spent fuel pool, the parameters validated would include heat-up rate of the pool; time to equilibrium temperature and variation of equilibrium temperature with time, especially after the two-phase mixture level drops below the top of fuel racks; temperature, pressure, and humidity in the environment above the pool; time at which liquid level drops below the fuel racks; time at which air–water vapor convective flow begins through fuel assemblies; heat-up rate of fuel rods; cladding failure location and size and melt relocation; and, if possible, radioactive material releases from the damaged fuel and their retention in the reactor building.

dation 4B (NRC, 2006) encourages the USNRC to upgrade the requirements of its spent fuel storage regulations to improve the resistance of dry casks to terrorist attacks.

- **Finding 4C** (NRC, 2006) notes that dry cask storage does not eliminate the need for pool storage at operating commercial reactors.
- **Finding 4D** (NRC, 2006) notes that dry cask storage for older, cooler spent fuel has inherent advantages over pool storage.
- **Finding 4E** (NRC, 2006) notes that the USNRC might determine that earlier-than-planned movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants, depending on the outcome of the analyses recommended in NRC (2004).

Findings 4C and 4D in NRC (2006) are statements of fact that require no reevaluation. Consequently, they are not discussed further. The other findings and recommendations are reevaluated in the followings sections.

4.3.1 Reevaluation of Finding 4A from NRC (2006)

Additional work is now being carried out by the USNRC to understand the robustness of dry casks to terrorist attacks. A general description of this work is given in a March 29, 2012, memo to the director of the USNRC's Office of Regulatory Research[14] and is to include physical testing as well as analysis and evaluation of existing studies. The present committee received a classified briefing on this work. These studies are addressing a range of attack scenarios and appear to be well conceived. However, because this work is still under way, the committee finds (**Finding 4.9**) that it is unable to assess the work's technical soundness and completeness.

4.3.2 Reevaluation of Finding and Recommendation 4B from NRC (2006)

The USNRC is incorporating the result of its analysis on dry cask vulnerabilities into its regulations through rulemaking. The rulemaking was still in progress[15] when the present study was being completed; consequently, the committee finds (**Finding 4.10**) that it is unable to evaluate its technical soundness and completeness. The committee recommends (**Recommendation 4.10**) that the USNRC should give high priority to com-

[14] Available at http://pbadupws.nrc.gov/docs/ML1203/ML120380260.pdf.

[15] On October 6, 2015, the Commission approved a 5-year delay in the commencement of this rulemaking. See http://pbadupws.nrc.gov/docs/ML1528/ML15280A105.pdf.

pleting these analyses and rulemaking. Additional information is provided in Appendix 4A.

4.3.3 Reevaluation of Finding 4E from NRC (2006)

The USNRC has completed technical and regulatory studies (USNRC, 2013, 2014a) to inform a regulatory decision on the need for earlier-than-planned movements (i.e., expedited transfer) of spent fuel at commercial nuclear plants from pools to dry cask storage. The technical study examined the consequences of a large earthquake on a spent fuel pool at a particular nuclear plant. Sensitivity analyses were performed to generalize the results of this study to spent fuel pools at other U.S. nuclear plants. The regulatory study included a safety goal screening and cost-benefit analysis to assess the benefit of expedited transfer of spent fuel from pools to dry casks. These USNRC analyses are described in more detail in Chapter 7 of this report.

The present committee finds (**Finding 4.11**) that this analysis did not consider spent fuel storage sabotage risks, dry cask storage risks, or certain health consequences that would likely result from a severe nuclear accident. The analysis also used simplifying bounding assumptions that make it technically difficult to assign confidence intervals to the consequence estimates or make valid risk comparisons. A risk assessment that evaluates the three questions of the risk triplet (see Chapter 5) and that accounts for uncertainties in both probability and consequence estimates is needed to address Finding 4E in NRC (2006) to determine whether "earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants." The committee recommends (**Recommendation 4.11**) that the USNRC should perform a spent fuel storage risk assessment to elucidate the risks and potential benefits of expedited transfer of spent fuel from pools to dry casks. Chapter 7 provides the committee's analysis to support these findings and recommendations.

4.4 IMPLEMENTATION ISSUES

Chapter 5 of NRC (2006) provides a discussion of potential impediments to implementing the recommendations in that report. The impediments involve the timely completion of the expert analyses and ensuring that the results of those analyses are communicated to the nuclear industry so that appropriate and timely mitigating actions can be taken. The chapter contains one finding and recommendation (see Table 4.1).

Finding 5A (NRC, 2006) notes that security restrictions on the sharing of information and analyses is hindering progress in addressing potential vulnerabilities of spent fuel storage to terrorist attacks. **Recommenda-**

tion 5A (NRC, 2006) encourages the USNRC to improve its sharing of information on the vulnerability and consequence studies with nuclear plant operators and dry cask storage vendors on a timely basis.

The present committee received briefings from USNRC staff and the Nuclear Energy Institute (NEI), which represents nuclear plant operators and dry cask storage vendors, on current practices for sharing security-related information. The committee finds (**Finding 4.12**) that the USNRC has made a commendable effort to improve the sharing of pertinent information on vulnerability and consequence analyses of spent fuel storage with nuclear power plant operators and dry cask storage system vendors. The Commission has sponsored key staff at these organizations for national security clearances, regularly shares important security-related information and threat-related intelligence[16] with industry groups, and is responsive to industry requests for information. An NEI representative informed the committee that the industry is satisfied with the content and timeliness of security-related information that it is receiving from the Commission.

[16] The USNRC receives intelligence information from other government agencies on a regular basis. This information is used to assess the threat environment and develop appropriate notifications to plant licensees.

TABLE 4.1 Committee Reevaluation of Findings and Recommendations from NRC (2006)

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
Terrorist Attacks on Spent Fuel Storage or Theft of Spent Fuel	
FINDING 2A: The probability of terrorist attacks on spent fuel storage cannot be assessed quantitatively or comparatively. Spent fuel storage facilities cannot be dismissed as targets for such attacks because it is not possible to predict the behavior and motivations of terrorists, and because of the attractiveness of spent fuel as a terrorist target given the well-known public dread of radiation.	**FINDING 4.1:** The understanding of security risks at nuclear power plants and spent fuel storage facilities can be improved through risk assessment. Event trees and other representational formalisms can be used to systematically explore terrorist attack scenarios, responses, and potential consequences. Expert elicitation can be used to rank scenarios; develop likelihood estimates; and characterize adaptive adversary responses to various preventive, protective, or deterrence actions. The identification of scenarios may be incomplete, and the estimates developed through expert elicitation are subjective and can have large uncertainties. Nevertheless, risk assessment methods that focus on the risk triplet—scenarios, likelihoods, and consequences—can contribute useful security insights. **RECOMMENDATION 4.1A:** The U.S. nuclear industry and the U.S. Nuclear Regulatory Commission should strengthen their capabilities for identifying, evaluating, and managing the risks from terrorist attacks. Particular attention is needed to broaden scenario identification, including asymmetric attacks; account for the adaptive nature of adversaries; account for the performance of plant security personnel in responding to the identified scenarios; estimate the potential onsite and offsite consequences of attack scenarios, including radioactive releases and psychological impacts; and develop strategies for countering the identified threats.

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
	RECOMMENDATION 4.1B: The U.S. Nuclear Regulatory Commission should sponsor a spent fuel storage (wet and dry storage) security risk assessment for U.S. nuclear plants. The primary objectives of this assessment should be to (1) develop and exercise the appropriate methodologies for characterizing risk and estimating uncertainties, and (2) explore the benefits of risk assessment for enhancing security at U.S. nuclear plants. This assessment should be subjected to independent review by technical peers (i.e., peer review) as part of the development process.
FINDING 2B: The committee judges that the likelihood terrorists could steal enough spent fuel for use in a significant radiological dispersal device is small. **RECOMMENDATION (2B):** The Nuclear Regulatory Commission should review and upgrade, where necessary, its security requirements for protecting spent fuel rods not contained in fuel assemblies from theft by knowledgeable insiders, especially in facilities where individual fuel rods or portions of rods are being stored in pools.	**FINDING 4.2:** The U.S. Nuclear Regulatory Commission has made good progress in upgrading its requirements for protecting spent fuel rods not contained in assemblies. The Commission has taken actions to improve inventory controls, enhance inspections, and update regulatory guidance for control and accounting of spent fuel rods and rod fragments. The Commission is also undertaking a rulemaking to clarify and strengthen material control and accounting requirements for these materials.

continued

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
FINDING 2C: A number of security improvements at nuclear power plants have been instituted since the events of September 11, 2001. The Nuclear Regulatory Commission did not provide the committee with enough information to evaluate the effectiveness of these procedures for protecting stored spent fuel. **RECOMMENDATION (2C):** Although the committee did not specifically investigate the effectiveness and adequacy of improved surveillance and security measures for protecting stored spent fuel, an assessment of current measures should be performed by an independent* organization. (*That is, independent of the Nuclear Regulatory Commission and the nuclear industry.)	**FINDING 4.3:** The U.S. Nuclear Regulatory Commission has not carried out an independent examination of surveillance and security measures for protecting stored spent fuel that was recommended by NRC (2006). **RECOMMENDATION 4.3:** The independent examination of surveillance and security measures for protecting stored spent fuel recommended by NRC (2006) should include an examination of the effectiveness of the U.S. Nuclear Regulatory Commission's programs for mitigating insider threats.
Safety and Security of Pool Storage	
FINDING 3A: Pool storage is required at all operating commercial nuclear power plants to cool newly discharged spent fuel.	No change.
FINDING 3B: The committee finds that, under some conditions, a terrorist attack that partially or completely drained a spent fuel pool could lead to a propagating zirconium cladding fire and the release of large quantities of radioactive materials to the environment. Details are provided in the committee's classified report.	**FINDING 4.4:** The present committee does not have a sufficient information to reevaluate Finding 3B in NRC (2004) because the U.S. Nuclear Regulatory Commission has not examined the specific terrorist attack scenarios identified in that report (see Recommendation 3E-1 in NRC [2004]).

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	**Present committee's reevaluation of findings and recommendations in NRC** (2006)
FINDING 3C: It appears to be feasible to reduce the likelihood of a zirconium cladding fire following a loss-of-pool-coolant event using readily implemented measures.	**FINDING 4.5:** Technical analyses undertaken by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories after 2004 confirm that reconfiguring spent fuel in pools can be an effective strategy for reducing the likelihood of fuel damage and zirconium cladding fires following loss-of-pool-coolant events. However, reconfiguring spent fuel in pools does not eliminate the risks of zirconium cladding fires, particularly during certain periods following reactor shutdowns or for certain types of pool drainage conditions. These technical studies also illustrate the importance of maintaining water coolant levels in spent fuel pools so that fuel assemblies do not become uncovered.
FINDING 3D: The potential vulnerabilities of spent fuel pools to terrorist attacks are plant-design specific. Therefore, specific vulnerabilities can be understood only by examining the characteristics of spent fuel storage at each plant.	No change.

continued

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	**Present committee's reevaluation of findings and recommendations in NRC (2006)**
FINDING 3E: The Nuclear Regulatory Commission and independent analysts have made progress in understanding some vulnerabilities of spent fuel pools to certain terrorist attacks and the consequences of such attacks for releases of radioactivity to the environment. However, additional work on specific issues is needed urgently. **RECOMMENDATION (3E-1):** The Nuclear Regulatory Commission should undertake additional best-estimate analyses to more fully understand the vulnerabilities and consequences of loss-of-pool-coolant events that could lead to a zirconium cladding fire. Based on these analyses, the Commission should take appropriate actions to address any significant vulnerabilities that are identified. The analyses of the BWR and PWR spent fuel pools should be extended to consider the consequences of loss-of-pool-coolant events that are described in the committee's classified report. The consequence analyses should address the following questions: • To what extent would such attacks damage the spent fuel in the pool, and what would be the thermal consequences of such damage? • Is it feasible to reconfigure the spent fuel within pools to prevent zirconium cladding fires given the actual characteristics (i.e., heat generation) of spent fuel assemblies in the pool, even if the fuel were damaged in an attack? Is there enough space in the pools at all commercial reactor sites to implement such fuel reconfiguration? • In the event of a localized zirconium cladding fire, will such rearrangement prevent its spread to the rest of the pool? • How much spray cooling is needed to prevent zirconium cladding fires and prevent propagation of such fires? Which of the different options for providing spray cooling are effective under attack and accident conditions?	**FINDING 4.6:** Additional analyses and physical experiments carried out by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories since NRC (2006) was completed have substantially improved the state of knowledge of boiling water reactor (BWR) and pressurized water reactor (PWR) spent fuel behavior following partial or complete loss of pool water. These studies and experiments have addressed the following important issues: • Fuel damage state and timing as a function of fuel age and pool water loss. • Propagation of zirconium cladding fires to other assemblies in the pool. • Potential mitigation strategies (dispersion of hot fuel assemblies in the pool, water sprays, water replacement) for delaying or preventing fuel damage following pool water loss. These experiments have resulted in significant validation of the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code that is used to model coolant loss in spent fuel pools. However, the code is unable to adequately model flows when stratification occurs and plumes form in the pool and/or above-pool environment. Moreover, key portions of code lack validation, and there has been no end-to-end validation of the code for modeling coolant loss in spent fuel pools. **RECOMMENDATION 4.6** The U.S. Nuclear Regulatory Commission should (1) sponsor an end-to-end validation of the MELCOR code for use in modeling coolant loss in spent fuel pools, and (2) validate key submodels in the code with particular attention paid to • Modeling the thermal and chemical behavior of spent fuel assemblies in partially drained pools. • Modeling the thermal and chemical response of spent fuel assemblies to the application of water sprays. • Modeling and validating for stratified flows in fully and partially drained pools.

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
Sensitivity analyses should also be undertaken to account for the full range of variation in spent fuel pool designs (e.g., rack designs, capacities, spent fuel burn-ups, and ages) at U.S. commercial nuclear power plants. **RECOMMENDATION (3E-2):** While the work described in the previous recommendation under Finding 3E, above, is being carried out, the Nuclear Regulatory Commission should ensure that power plant operators take prompt and effective measures to reduce the consequences of loss-of-pool-coolant events in spent fuel pools that could result in propagating zirconium cladding fires. The committee judges that there are at least two such measures that should be implemented promptly: • Reconfiguring of fuel in the pools so that high decay-heat fuel assemblies are surrounded by low decay-heat assemblies. This will more evenly distribute decay-heat loads, thus enhancing radiative heat transfer in the event of a loss of pool coolant. • Provision for water-spray systems that would be able to cool the fuel even if the pool or overlying building were severely damaged.	**FINDING 4.7:** The U.S. Nuclear Regulatory Commission has not analyzed the vulnerabilities of spent fuel pools to the specific terrorist attack scenarios identified in Recommendation 3E-1 in NRC (2004). **FINDING 4.8:** The U.S. Nuclear Regulatory Commission and the U.S. nuclear industry have made good progress in implementing actions to address Recommendation 3E-2 in NRC (2006). The U.S. Nuclear Regulatory Commission has directed plant licensees to • Reconfigure their spent fuel in pools to achieve at least a 1 × 4 dispersion of high- and low-decay-heat assemblies, unless such configuration can be shown to be inapplicable or unachievable. This configuration must be achieved following each fuel offload from the reactor not later than 60 days after reactor shutdown. • Develop guidance and implement strategies to maintain and restore spent fuel pool cooling following explosions and fires. To address this requirement, the U.S. nuclear industry has developed and adopted guidance and strategies for spent fuel pool water makeup and water sprays. However, additional work is needed to more fully implement Recommendation 3E-2 in NRC (2006). **RECOMMENDATION 4.8:** The U.S. Nuclear Regulatory Commission should take the following actions to more fully implement Recommendation 3E-2 in NRC (2006): • Reexamine the need for the 60-day limit for fuel dispersion and reduce the allowable time if feasible. • Reexamine and, if needed, redesign the water makeup and spray systems and strategies to ensure that they can be implemented when physical access to pools is hindered or the site becomes inaccessible.

continued

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
Safety and Security of Dry Cask Storage and Comparison with Pool Storage	
FINDING 4A: Although there are differences in the robustness of different dry cask designs (e.g., bare fuel versus canister based), the differences are not large when measured by the absolute magnitudes of radionuclide releases in the event of a breach.	**FINDING 4.9:** Additional analyses on dry cask vulnerabilities have been undertaken since NRC (2006) was completed. This work was still under way when the present report was being completed; consequently, the committee was unable to evaluate its technical soundness and completeness.
FINDING 4B: Additional steps can be taken to make dry casks less vulnerable to potential terrorist attacks. Note: Section 4.2.3 in the unclassified (2006) report lists such steps: Adding additional surveillance; adding berms; adding visual barriers; increasing spacing between casks or adding shims; making relatively minor changes in cask designs. **RECOMMENDATION (4B):** The Nuclear Regulatory Commission should consider using the results of the vulnerability analyses for possible upgrades of requirements in 10 CFR 72 for dry casks, specifically to improve their resistance to terrorist attacks.	**FINDING 4.10:** The U.S. Nuclear Regulatory Commission is incorporating the results of its dry cask vulnerability analyses into its regulations through rulemaking. The rulemaking was still in progress when the present study was being completed; consequently, the committee was unable to evaluate its technical soundness and completeness. **RECOMMENDATION 4.10:** The U.S. Nuclear Regulatory Commission should give high priority to completing its analyses on dry cask storage vulnerabilities and rulemaking.
FINDING 4C: Dry cask storage does not eliminate the need for pool storage at operating commercial reactors.	No change.
FINDING 4D: Dry cask storage for older, cooler spent fuel has two inherent advantages over pool storage: (1) It is a passive system that relies on natural air circulation for cooling, and (2) it divides the inventory of that spent fuel among a large number of discrete, robust containers. These factors make it more difficult to attack a large amount of spent fuel at one time and also reduce the consequences of such attacks.	No change.

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
FINDING 4E: Depending on the outcome of plant-specific vulnerability analyses described in the committee's classified report, the Nuclear Regulatory Commission might determine that earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants.	**FINDING 4.11:** The U.S. Nuclear Regulatory Commission has completed a technical analysis of spent fuel pool accident consequences to inform a regulatory decision on expedited transfer of spent fuel from pool to dry cask storage. The analysis was carried out in accordance with prescribed U.S. Nuclear Regulatory Commission regulatory guidance and provides valuable technical information about the impacts of various accident scenarios on spent fuel storage in pools. However, the analysis did not consider spent fuel storage sabotage risks, dry cask storage risks, or certain health consequences that would likely result from a severe nuclear accident. The analysis also used simplifying bounding assumptions that make it technically difficult to assign confidence intervals to the consequence estimates or make valid risk comparisons. A risk assessment that evaluates the three questions of the risk triplet and that accounts for uncertainties in both probability and consequence estimates is needed to address Finding 4E in NRC (2006) to determine whether "earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants." **RECOMMENDATION 4.11:** The U.S. Nuclear Regulatory Commission should perform a spent fuel storage risk assessment to elucidate the risks and potential benefits of expedited transfer of spent fuel from pools to dry casks. This risk assessment should address accident and sabotage risks for both pool and dry storage. The sabotage risks should be assessed using the methodology developed in response to the present committee's Recommendation 4.1B.

continued

TABLE 4.1 Continued

Findings and recommendations from NRC (2006)	Present committee's reevaluation of findings and recommendations in NRC (2006)
Implementation Issues	
FINDING 5A: Security restrictions on sharing of information and analyses are hindering progress in addressing potential vulnerabilities of spent fuel storage to terrorist attacks. **RECOMMENDATION (5A):** The Nuclear Regulatory Commission should improve the sharing of pertinent information on vulnerability and consequence analyses of spent fuel storage with nuclear power plant operators and dry cask storage system vendors on a timely basis.	**FINDING 4.12:** The U.S. Nuclear Regulatory Commission has made a commendable effort to improve the sharing of pertinent information on vulnerability and consequence analyses of spent fuel storage with nuclear power plant operators and dry cask storage system vendors. The Commission has sponsored key staff at these organizations for national security clearances, regularly shares important security-related information and threat-related intelligence with industry groups, and is responsive to industry requests for information.

APPENDIX 4A
Dry Cask Storage Regulations

There are two types of USNRC licenses for dry storage of spent fuel:

1. A site-specific license issued under 10 CFR Part 72 (*Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*) for an Independent Spent Fuel Storage Installation[1] (ISFSI) either on- or offsite, and
2. A general license granted under 10 CFR Part 72 to 10 CFR Part 50 (*Domestic Licensing of Production and Utilization Facilities*) licensees under certain conditions.

A site-specific license requires an application, with supporting documentation, which upon approval culminates in a license issued to the installation. A general license is granted by the regulations if the installation meets certain conditions. There were 14 sites that dry-store spent fuel under site-specific licenses[2] and 60 sites that dry-store spent fuel under a general license in the United States as of August 13, 2015.[3]

Security requirements under each type of license can vary depending on whether the storage facility is (1) co-located with an operating reactor, (2) co-located with a decommissioned reactor, or (3) located offsite. Generally, the first two types of storage facilities may be generally or specifically licensed but the third type must be specifically licensed; consequently, there are three different sets of requirements found in regulation, license conditions, and emergency orders enforced by the USNRC (see Figure 4A.1).

Emergency response planning requirements for dry storage facilities may also vary depending on whether the reactor is operational or decommissioned and whether the facility is located on- or offsite. Location-based (i.e., facility located on- or offsite) variations in planning requirements are generally regulation driven, whereas reactor status-based (i.e., reactor operating or decommissioned) variations are usually exemption driven. Licensees that dry-store spent fuel at generally licensed sites that are co-located with a reactor may request exemptions from certain emergency planning

[1] An ISFSI is a facility used to store spent nuclear fuel and certain other types of radioactive material (e.g., greater-than-class-C low-level waste) on an interim basis. The USNRC considers an ISFSI to be “independent” even when it is co-located with another USNRC-licensed facility such as a power reactor.

[2] Additionally, one site (GE Morris, located near Chicago, Illinois) operates a pool-storage facility under a site-specific license.

[3] See http://pbadupws.nrc.gov/docs/ML1524/ML15240A058.pdf.

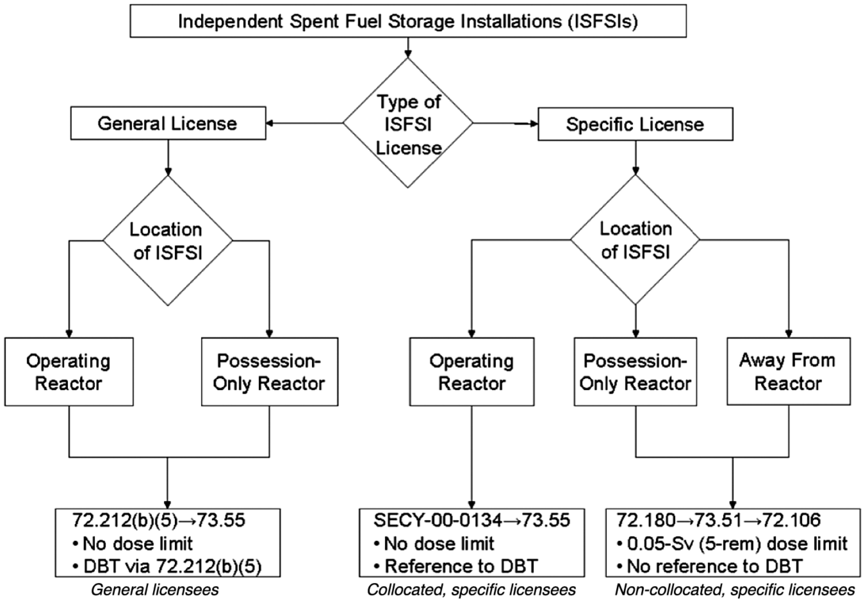

FIGURE 4A.1 Current USNRC security regulations for ISFSIs. SOURCE: USNRC (2009a, Figure 1).

requirements once there is a permanent cessation of operations and fuel has been removed from the reactor vessel.[4]

Owing to this complicated regulatory scheme, it is not a simple matter to assess the overall security and safety of ISFSIs. A complete assessment would require a case-by-case evaluation of each facility under its applicable requirements, physical configuration, and other site-specific information. The USNRC has pointed out that "continuing differences between general-license and specific license ISFSI security requirements is not appropriate and does not contribute to long term regulatory stability or to stakeholder support and understanding of the Commission's regulatory programs for storing spent fuel" (USNRC, 2007a, Enclosure 3, p. 9).

Following the September 11, 2001, terrorist attacks on the United States, the USNRC carried out an evaluation of its regulatory program for spent fuel storage and initiated a rulemaking (USNRC, 2009a) "to create a more consistent and coherent regulatory structure for these types of waste storage facilities." The USNRC decided to undertake a new security

[4] NSIR/DPR-ISG-02, Interim Staff Guidance, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants.

rulemaking that is risk informed and performance based. "Risk informed" refers to a vulnerability assessment methodology that takes into consideration both threat- and non-threat-based information. "Performance based" refers to the application of a dose acceptance limit to ISFSIs. According to the USNRC, radiological sabotage scenarios would be developed, and ISFSIs would be expected to provide high assurance that a 5-rem lifetime dose to the maximally exposed individual would not be exceeded at the facility boundary.

The security requirements are expected to be captured in a regulatory guidance document. USNRC staff will develop these security requirements from a vulnerability perspective (not a threat perspective). Based on this guidance, ISFSIs would be subjected to an analysis to determine whether they meet the dose-limit requirements. All ISFSIs would be held to the same standard of protection regardless of whether they are licensed under 10 CFR Part 50 or 10 CFR Part 72. The USNRC judges that this approach would be relatively simple for current ISFSIs to satisfy (assuming that they now can meet the 5-rem standard) but would move away from the design-basis-threat requirement that now applies to most ISFSIs (USNRC, 2007a, Enclosure 3).

At the time the present report was being written, the USNRC's ISFSI security rulemaking actions had not been completed, and the future of this rulemaking is uncertain. On October 6, 2015, the Commission approved a 5-year delay in the commencement of this rulemaking.[5] In the memorandum that discusses this decision, the USNRC staff notes that this rulemaking could be accelerated under certain circumstances. It also notes that at the end of this 5-year period "the staff should re-evaluate whether rulemaking in this area is warranted." Therefore, it is premature for the present committee to comment on possible changes to the regulatory framework. Nevertheless, the committee recommends (**Recommendation 4.10**) that the USNRC give high priority to completing its analyses on dry cask storage vulnerabilities and rulemaking.

[5] See http://pbadupws.nrc.gov/docs/ML1528/ML15280A105.pdf.

5

Security Risk Assessment

This chapter provides supporting information for the present committee's **Finding 4.1** and **Recommendations 4.1A** and **4.1B** in Chapter 4 (Table 4.1):

FINDING 4.1: The understanding of security risks at nuclear power plants and spent fuel storage facilities can be improved through risk assessment. Event trees and other representational formalisms can be used to systematically explore terrorist attack scenarios, responses, and potential consequences. Expert elicitation can be used to rank scenarios; develop likelihood estimates; and characterize adaptive adversary responses to various preventive, protective, or deterrence actions. The identification of scenarios may be incomplete, and the estimates developed through expert elicitation are subjective and can have large uncertainties. Nevertheless, risk assessment methods that focus on the risk triplet—scenarios, likelihoods, and consequences—can contribute useful security insights.

RECOMMENDATION 4.1A: The U.S. nuclear industry and the U.S. Nuclear Regulatory Commission should strengthen their capabilities for identifying, evaluating, and managing the risks from terrorist attacks. Particular attention is needed to broaden scenario identification, including asymmetric attacks; account for the adaptive nature of adversaries; account for the performance of plant security personnel in responding to the identified scenarios; estimate the potential onsite and offsite consequences of attack scenarios, including radioactive releases

SIDEBAR 5.1
Hazard, Threat, and Risk

The terms *hazard*, *threat*, and *risk* are used throughout this report. These terms are related but not interchangeable.

The terms *hazard* and *threat* are used to denote conditions that have the potential to cause harm or other adverse effects. Hazard is usually used in a safety context to denote conditions that can result in accidents, whereas threat is usually used in a security context to denote potential malevolent acts.

The term *risk* can be defined in terms of a triplet of questions:

1. What can go wrong?
2. How likely is that to happen?
3. What are the consequences if it does happen?

Hazards and threats are related to the first question of the risk triplet; that is, "What can go wrong?" in a safety or security context.

and psychological impacts; and develop strategies for countering the identified threats.

RECOMMENDATION 4.1B: The U.S. Nuclear Regulatory Commission should sponsor a spent fuel storage (wet and dry storage) security risk assessment for U.S. nuclear plants. The primary objectives of this assessment should be to (1) develop and exercise the appropriate methodologies for characterizing risk and estimating uncertainties, and (2) explore the benefits of risk assessment for enhancing security at U.S. nuclear plants. This assessment should be subjected to independent review by technical peers (i.e., peer review) as part of the development process.

Sidebar 5.1 provides definitions for some terms that are used in this chapter.

5.1 BACKGROUND ON RISK ASSESSMENT

Risk assessment is a formalized thought process for answering the following triplet of questions (Kaplan and Garrick, 1981):

1. What can go wrong?
2. How likely is that to happen?
3. What are the consequences if it does happen?

Probabilistic risk assessment (PRA) is a highly developed methodology for performing a risk assessment that is widely applied by the nuclear industry and its regulator. The adjective "probabilistic" is included in this terminology to emphasize that the likelihood (second item) of an event is expressed in the assessment. As noted in Appendix I in the committee's phase 1 report (NRC, 2014), PRA describes the application of risk assessment to accidents at nuclear plants. In the following, security risk assessment will refer to assessments in which the likelihood of terrorist events is included in the evaluation process.

The specific metrics chosen to express the risk depend on the system or activity to be informed by the risk assessment. Crucial to all modern risk assessments (e.g., USNRC, 1990) is the recognition that their results are uncertain and that this uncertainty needs to be reflected in the results. Results are typically presented in terms of uncertainty distributions rather than point values (e.g., EPRI, 2012b; USNRC, 2013). The U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.200 (USNRC, 2009b) and the PRA consensus standard published jointly by the American Nuclear Society and the American Society for Mechanical Engineers (ASME/ANS, 2009) emphasize the importance of identifying and understanding uncertainties for achieving technical acceptability in a PRA.

5.2 APPLICATION OF RISK ASSESSMENT TO SECURITY

The identification of terrorist threats against reactors and spent fuel pools is a necessary part of security planning at all nuclear plants (Sidebar 5.2). Analyses or exercises can be undertaken for each identified threat to explore whether the terrorist is likely to succeed in causing significant damage, and defenses can be adjusted accordingly. But whether the identified set of threats is complete is generally unknown. As discussed in Chapter 3, there is also a pressing need to more systematically identify potential cyber, insider, and asymmetric[1] security threats. More formalized processes for identifying and analyzing threats—for example PRA—could help to improve security at nuclear plants.

The National Research Council public report (NRC, 2006) questioned the feasibility of applying PRA to nuclear plant security because of the difficulty of developing a complete set of bounding attack scenarios and estimating their likelihoods of occurrence. The report noted correctly that attack probabilities depend on factors such as terrorist motives, expertise, and access to technical means, which may be difficult or impossible to

[1] *Asymmetric attacks* refer to attacks where there are dissimilarities in the capabilities, strategies, and/or tactics between an adversary and a defending force. Additional discussion of asymmetric attacks is provided in Chapter 3.

SIDEBAR 5.2
Hazard Assessments in Technological Systems

Hazard assessments in technological systems have evolved through three stages as knowledge about system design and performance has improved and analytical sophistication has increased. These stages and their applications are described in this sidebar.

The first stage of a hazard assessment involves the formulation of *design-basis hazards* that must be considered in the development of a technology's safety or security systems. Hazard formulation is most effective when a conceptual design of the technological system is available.

The second stage involves the performance of a *system-wide risk assessment*, which integrates the contributions of individual components of a technological system to estimate overall system performance. The use of risk assessment permits the influences of uncertainties and sensitivities in system performance to be made explicit. It also permits exploration of beyond-design-basis hazards and their contribution to overall risk.

The third stage involves the development of a refined set of design-basis hazards using the system-wide risk assessment described above. This allows for the creation of *decision rules* for improving the efficiency of decision making and resource use.

Hazard assessments in many technological systems do not progress beyond stage 1, depending on the policies of the particular industry and demands of society. In cases of increased social concern—for example, nuclear power, commercial aviation, and aerospace applications—stage 2 approaches are used routinely for assessing safety hazards. Stage 3 approaches are also used in some specialized nuclear power applications, for example in the development of the USNRC's Maintenance Rule and Reactor Oversight Process.

Security hazard assessments in nuclear power applications generally utilize stage 1 approaches. Their focus has been on design-basis threats. However, as discussed in this chapter, there are no technical obstacles to applying stage 2 and 3 approaches to these assessments. Risk assessment can be used to explore the contribution of beyond-design-basis threats to overall risk.

know. Although NRC (2006) expressed reservations about the possibility of quantifying risk, it also indicated that qualitative judgments could be made about the relative vulnerabilities of spent fuel storage facilities to various terrorist attack scenarios described in the classified report (NRC, 2004).

The present committee agrees with NRC (2006) that there are technical challenges associated with identifying terrorist attack scenarios and quantifying their likelihoods. Nevertheless, the committee judges that the risks of terrorist attacks on nuclear plants and spent fuel storage facilities can be characterized by adapting well-established risk assessment methods. For example,

- Event trees and other representational formalisms can be used to systematically explore terrorist attack scenarios[2] and their potential consequences. One can, for example, generate event trees with associated likelihoods that represent "baseline" conditions at a nuclear plant at the beginning of a terrorist attack scenario. These event trees and their associated likelihoods can be modified as the scenario unfolds to reflect defensive actions by plant personnel and terrorist adjustments to those actions.
- These scenarios and their associated likelihoods (usually expressed as frequency distributions) can be developed and estimated, respectively, using established methods such as expert elicitation and CARVER analysis (see Appendixes 5A and 5B). It may be possible to use likelihood estimates to rank scenarios and identify their comparative importance if their uncertainties are small and/or uncorrelated.

The committee recognizes, of course, that the set of scenarios identified using such methods will likely be incomplete, and likelihood estimates and scenario rankings will be subjective and may have large uncertainties. Nevertheless, the use of these methods provides greater technical rigor and transparency to an analysis than traditional deterministic methods such as the design-basis threat (DBT) (see Chapter 3).

The use of risk assessment can help to

- Broaden scenario identification for both physical and cyber terrorist attacks, including insider and asymmetric attacks;
- Account for the performance of plant security personnel in responding to the identified scenarios;
- Identify potential onsite and offsite consequences of such scenarios, ranging from radioactive releases to psychological impacts; and
- Characterize uncertainties in the scenarios, likelihoods, and consequences.

In fact, risk assessment can provide useful security insights that are analogous to the insights derived from safety risk assessments.

Risk assessment allows for the orderly development of conclusions that reflect the totality of information available about a system's performance in particular circumstances. Such assessments can provide evaluations of risks in terms of the frequencies of occurrence of random events or consequences conditional on occurrence of a particular event such as a terrorist attack. Both types of assessments can be valuable for identifying the relative impor-

[2] Such a scenario would describe the characteristics of the attack including the size and weaponry of the attacking force, its tactics, and the plant systems that are targeted.

tance of the various contributors to performance outcomes and the more sensitive elements of the system in affecting such outcomes. For either type of risk assessment, explicit treatments of uncertainty are essential, because the components of the assessment (e.g., scenarios, likelihoods, and consequences) can have substantial uncertainties.[3]

The adaptation of PRA to security would establish a common framework for assessing risks at nuclear plants. This would provide a consistent basis for operational and regulatory decision making about risks, including at the safety-security interface; it could provide further opportunities to risk-inform security regulations; and it could help improve cost-benefit analyses associated with the backfit rule (see Chapter 5 in NRC [2014]).

The current state of development of risk analysis for nuclear plant security is similar to that for safety risk analysis in the early 1970s. At that time it was argued that characterizing the likelihoods of physical accidents was infeasible or at least impractical because they had such low probabilities of occurrence and large uncertainties. There has been considerable technical progress in the use of risk assessment for nuclear plant safety[4] over the past four decades. The committee judges that it is not unreasonable to expect that similar progress can be achieved with security risk assessment. In fact, efforts are already under way to further develop this methodology.

The USNRC sponsored a Risk Informed Security Regulation Workshop[5] in 2014 to discuss the current state of efforts to use risk assessment in nuclear plant security and to identify opportunities to risk-inform security regulations. The workshop identified sabotage-initiating-event definition and uncertainty estimation as key areas for further development, but it did not recommend explicit actions to achieve such development. It did, however, suggest that a security risk assessment effort be undertaken.

In a keynote address at the workshop, then-Commissioner George Apostolakis argued that the DBT paradigm for nuclear plant security was too restrictive, and he discussed the usefulness of the expert judgement elicitation-based approach for characterizing threats and for integrating safety and security assessments. He noted that a common framework for safety and security would enable consistent decision making and the explicit treatment of the safety and security interface.

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[3] Presentation and propagation of uncertainties is particularly important in the case of security risk assessment because the uncertainties associated with initiating threat events and consequences are likely to be much larger than for natural events and accidents. Consequently, reliance on measures of central tendency may be particularly misleading.

[4] See Appendix I of the present committee's phase 1 report (NRC, 2014) for a discussion of the history and current state of risk assessment practice as well as needs for further technical advancement.

[5] See http://www.inmm.org/AM/Template.cfm?Section=Risk_Informed_Security_Workshop1&Template=/CM/ContentDisplay.cfm&ContentID=4601.

5.3 SECURITY RISK ASSESSMENT METHODOLOGIES

Progress is currently being made in adapting and extending risk analysis to security applications. Procedures for conducting security risk assessments do not yet have consensus-level agreement from professional standards organizations as do safety risk assessments, but there do exist "how-to, step-by-step" methods for performing security risk assessments (see, for example, EPRI, 2004; Garrick et al., 2004; Hirschberg et al., 2016). These methods were developed after the September 11, 2001, terrorist attacks.

Garrick et al. (2004) use the risk triplet as the organizing principle for conducting a security risk assessment for terrorist attacks that could have catastrophic consequences. They note that "there is an urgent need for (1) understanding the threats involved, (2) appreciating vulnerabilities, and (3) an analytical process for assessing the risk and mitigating the threat" (p. 131). A major contribution of this paper is a procedure for quantitative risk assessment of threats.

Their methodology involves three major steps: (1) analyzing and quantitatively assessing the threats, (2) characterizing the success state of the system under attack, and (3) performing a quantitative vulnerability assessment in which the threat analysis generates the initiating events for the vulnerability assessment. Garrick et al. (2004) remark that "initiating events are application dependent and require extensive involvement of experts—those who develop and analyze intelligence and those who are expert in the nature of the threat …" (p. 136). They illustrate and discuss their methodology with examples from various large critical infrastructures that need to be protected against terrorist threats.

The Electric Power Research Institute (EPRI) methodology uses the following definition for risk:

Risk = frequency of threat occurrence × probability of threat success (given the threat occurrence) × probability of consequence.

The frequency of threat occurrence is the most challenging factor to estimate quantitatively. EPRI suggests that the frequency of large-scale terrorist threat scenarios can be estimated by leveraging the national terrorism experience base. In the EPRI study, considerations in estimated frequency of terrorist threat included the following:

- Large-scale threats require significant planning and preparation to execute, so they have an annual frequency of occurrence in the United States of less than 1.0 per year.
- There are many potential high-profile targets in the United States other than nuclear plants.

- If nuclear plants are chosen as a target, only 1 out of the 103 (then) operating nuclear reactors would be expected to be attacked.

As an outgrowth of the EPRI study, the U.S. Department of Homeland Security (DHS) supported a more simplified, semiquantitative conditional risk assessment process for particular threats of interest to them; this process is referred to as the Risk Assessment and Management for Critical Asset Protection (RAMCAP) process.[6] RAMCAP was used to support an industry-led assessment of risks at each U.S. nuclear plant including reactors, spent fuel pools, and dry cask storage. Tabletop assessments were conducted for a spectrum of postulated security threats. Insights gained from the analysis included the following:

- Important risk scenarios are site specific.
- Compliance with the design-basis threat does not necessarily ensure negligible risk.
- Modest changes in design and/or procedures can make the facility more resistant to security threats and reduce health and/or economic consequences due to security threats.

Hirschberg et al. (2016) proposed an analytic approach that leverages intelligence community knowledge to derive quantitative risk estimates for terrorist threats with potential for catastrophic consequences. Their estimates are based on three elements:

1. Probability that an attack is conducted. This estimate is derived based on historical evidence of attractiveness of the target and evidence of terrorist activity in the country of interest.
2. Probability that a given terrorist scenario can be successfully implemented. This estimate is based on assessments of the required resources, time, know-how, and countermeasures in place.
3. The consequences of an attack in terms of fatalities, injuries, and land contamination.

The approach enables diverse sources of knowledge to be integrated within a common framework to generate a more complete picture of the likelihood of a successfully executed attack and the resulting consequences. Sources of knowledge include expertise from the political sciences and intelligence communities on the motivations of terrorists; knowledge from the military and security communities on scenario planning; and physical assessments of the performance of the engineered systems to derive consequences.

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[6] See http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=1592595.

The authors acknowledged that the risks associated with natural and engineered systems can be assessed with higher confidence than the motivations and actions of terrorists. Nevertheless, the analyses can provide useful insights in spite of the large variation in uncertainty. For example, the graphs (e.g., Hirschberg et al., 2016, Figure 12) presented by the authors indicate that there is considerable variability in predicted risk and suggest that the terrorist threat should not be neglected compared to accident risks for both hydro and nuclear facilities.

There have been a number of other approaches to security risk analysis including work by Clauset et al. (2007), Willis et al. (2007), and Willis and LaTourrette (2008).

5.4 CHALLENGES FOR APPLYING RISK ASSESSMENT TO SECURITY

The usefulness of PRA for assessing security risks has been discussed and debated at length in the technical literature, including in reports from the NRC.[7] One common line of argument against the application of PRA to security has to do with the lack of knowledge of adversaries and their capabilities, motivations, and strategies. This creates challenges for developing a complete set of attack scenarios as well as estimating attack probabilities.

For example, the NRC's Committee on Risk-Based Approaches for Securing the DOE Weapons Complex (NRC, 2011) expressed reservations with respect to quantification of risk largely because of difficulties in defining attack strategies that adversaries might employ and their success probabilities. The committee did, however, note that some of the tools and techniques associated with risk assessment, particularly the structured thinking process, could be useful for developing a comprehensive "systems" approach to security.

Attack probabilities are widely acknowledged to be the most challenging to estimate because they require knowledge, data, or modeling of the motivations, capabilities, and intentions of terrorists. All such estimates will benefit from guidance from knowledgeable experts, for example, members of the intelligence community who have the appropriate personnel security clearances to access sensitive national security information on terrorist threats. In light of the paucity of historical data[8] for terrorist attacks on

[7] Articles by Ezell et al. (2010) and Brown and Cox (2011a,b) provide good discussions of the advantages and disadvantages of applying PRA to security.

[8] Two publicly accessible data sources are the Global Terrorism Database (GTD) (http://www.start.umd.edu/gtd/) and the RAND Database on Worldwide Terrorism (http://www.rand.org/nsrd/projects/terrorism-incidents.html). About 0.02 percent of the entries (26 out of 140,000 in the GTD) make reference to commercial nuclear facilities. The committee did not examine classified or private databases on terrorism, sources that analysts with appropriate security clearances would be able to utilize.

commercial nuclear facilities, from which one might hope to derive attack frequencies, such reliance seems essential. Estimates of attack probabilities by experts can be provided as distributions. While the resulting estimates may have large uncertainties, they would be informed by the best available knowledge at the time of the analysis. Appendix 5A provides a discussion of expert elicitation methods.

The Committee to Review the Department of Homeland Security's Approach to Risk Analysis (NRC, 2010) expressed reservations regarding the possibility of conducting an all-hazards risk assessment that combines risks associated with natural hazards with security risks. However, the committee was more optimistic about using an integrated approach if the goal was to compare the benefits of multiple alternative options for reducing risks. They pointed out that an integrated analysis might illuminate options for simultaneously reducing the risks arising from natural hazards and terrorism.

NRC (2010) also recommended that "DHS should strengthen its scientific practices, such as documentation, validation, and peer review by technical experts external to DHS. This strengthening of its practices will also contribute greatly to the transparency of DHS's risk modeling and analysis. DHS should also bolster its internal capabilities in risk analysis as part of its upgrading of scientific practices" (p. 3).

Another line of argument against the application of PRA to security is that the probabilities associated with the likelihood of particular threats may shift in response to defensive actions. Terrorists, unlike natural hazards or engineered systems, are intelligent adaptive adversaries. The probability of an earthquake will remain fixed whether or not steps are taken to mitigate its consequences. However, the probability of a terrorist attack against a facility might change in response to protective or mitigative actions that make it a less attractive target.

This line of argument was made by the Committee on Methodological Improvements to the DHS's Biological Agent Risk Analysis (NRC, 2008). This committee reviewed DHS's tool[9] for assessing the risks associated with the intentional release of biological threat agents. The committee argued that terrorist threats, unlike natural hazards and engineered systems, are intelligent, goal-oriented, resourceful, and adaptive adversaries. Consequently, PRA methods that rely on static event trees and associated probabilities are not appropriate for modeling adversary strategy sets. The committee argued that DHS should use decision-oriented models "that explicitly recognize terrorists as intelligent adversaries who observe U.S. defensive preparations and seek to maximize achievement of their own objectives" (NRC, 2008, p. 3).

[9] The tool is referred to as the Biological Threat Risk Assessment.

Brown and Cox (2011b) further elaborated this line of argument. They noted the following:

> "The capacity of terrorists to seek and use information and to actively research different attack options before deciding what to do raises unique features of terrorism risk assessment that are not adequately addressed by conventional PRA for natural and engineered systems—in part because decisions based on such PRA estimates do not adequately hedge against the different probabilities that attackers may eventually act upon." (p. 196)

They argued that the very existence of a PRA that suggested differences in attack likelihoods might cause attackers to change their behavior, negating its value.[10]

Ezell et al. (2010) and Ezell and Collins (2011) addressed the challenges of modeling attack strategies of intelligent adversaries that were raised by NRC (2008) and Brown and Cox (2011b). They acknowledged the added complexity of modeling adaptive adversaries, but they also argued that one could develop a baseline of current terrorist motivations, intent, and capabilities and facility defenses and assess probabilities conditional on this baseline. In other words, event trees in a risk assessment can be thought of as a snapshot in time of threats, vulnerabilities, and consequences that are subject to change as an attack scenario progresses. Of course, this snapshot may be incomplete and can have large uncertainties. Once new defensive (i.e., preventative, protective, or deterrence) measures are introduced, event trees and their associated probabilities are reassessed and updated as needed.

5.5 DISCUSSION

The present committee is acutely aware that its Finding 4.1 is a substantial departure from previous conclusions of other NRC committees on the use of risk assessment in security applications. In developing this finding, the committee examined advances in risk assessment science and practice since the previous NRC spent fuel study (NRC, 2004, 2006) and deliberated on the potential for future advancements. Some of these advances are cited and discussed in this chapter. This finding is intended to encourage the nascent efforts by the USNRC and nuclear industry to develop security risk assessments—and also to encourage their further development and application by the broader risk assessment community.

The present committee recognizes that additional work will be required to further develop security risk assessment methodologies. Work is particu-

[10] Of course, security risk assessments contain security-related information and are therefore not publicly releasable.

larly needed to develop and exercise processes for estimating the elements of the risk triplet (i.e., scenarios, likelihoods, and consequences); estimating their uncertainties; and appropriately communicating results, including uncertainties, to decision makers. The usefulness of risk assessment for informing resource allocation, design and operational enhancement, and regulatory decisions will improve as these elements and associated tools are further developed.

Even at their present stage of development, however, security risk assessments can be useful for making relative comparisons of design or operational alternatives within a particular system/facility or between facilities—particularly when analyses are conducted by the same group of experts applying comparable assumptions. Such assessments could help to identify potential gaps in the current security frameworks and reveal vulnerabilities that are missed by conventional security assessments. See, for example, the present committee's Recommendation 4.11 (Chapter 7) for assessing the risks of storing spent fuel in pools versus dry casks.

The present committee concurs with Ezell et al.'s (2010) arguments about the usefulness of PRA for security assessments. One can construct event trees and assign probabilities based on expert judgement with the full understanding that base probabilities can change when different types of preventive, protective, or deterrence actions are considered. The identification of scenarios may be incomplete, and the probability estimates may have large uncertainties. Nevertheless, there is no fundamental technical limit to performing a quantitative analysis, even though the probability calculations will be more complicated and will need to account for adversary behavior.

The present committee also concurs with NRC (2010) about the usefulness of PRA for considering multiple alternative options for reducing risks. Moreover, that report's advice to DHS about strengthening its scientific practices for security risk assessment (see Section 5.4) is also applicable to the USNRC and the nuclear industry. Such strengthening can improve analytical rigor and transparency and help to advance the state of the art in risk modeling and analysis.

There is a variety of risk analysis frameworks that could be used to support terrorism risk analysis, either as inputs to a PRA or as additional complementary perspectives to support decision making. Most particularly, as recommended by the Committee reviewing the DHS Bioterrorism Risk Assessment (NRC, 2008), it would be advantageous to explore frameworks that model terrorists as intelligent adversaries. Potential approaches for representing the beliefs and behaviors of intelligent adversaries include use of decision trees, attack trees, Bayesian belief networks, game theory, and agent-based models (see Ezell et al. [2010] and Brown and Cox [2011a] for further discussion of alternative representational frameworks and models).

Terrorists, unlike natural hazards and engineered systems, are intelligent agents that will condition their behaviors on the behaviors of defenders. Thus, a major challenge for adapting PRA to security has been the quantification of adversary actions. Although there may be great uncertainty in how these actions evolve and, arguably, in the ability to quantitatively produce meaningful risk estimates based on these actions, the present committee nevertheless judges that it is worthwhile to explore security risks using relevant methods, concepts, and tools arising from the risk triplet framework as an important adjunct to the conventional approaches that are now used by the U.S. nuclear industry and its regulator.

Sidebar 7.4 in Chapter 7 illustrates the added complexities that arise when considering intelligent adversaries versus natural hazards or engineered systems. Earthquakes are just as likely to occur during any operating cycle of a nuclear plant, but terrorist attacks may be most likely to occur during certain operating cycles. Security risk assessments would need to recognize, represent, and numerically propagate this added level of behavioral complexity.

Much remains to be learned about the effectiveness of deterrent or delaying actions as well as the potential consequences should an attack succeed. Quantitative evaluations, however crude, could help the nuclear industry and its regulator develop strategies for preventing and/or mitigating terrorist attacks.

The continued expression of terrorist threats in society, including cyber and insider threats, underscores the need to develop improved approaches for understanding, preventing, and mitigating them, particularly threats directed against civilian nuclear facilities. Indeed, it would be imprudent not to consider the potential benefits of risk assessment, which has served to advance understanding and management of safety risks, to nuclear plant security. Moreover, only by developing and testing risk assessments through specific applications will its limitations become more fully understood and improvements to overcome them will be made.

APPENDIX 5A
Expert Elicitation

Expert elicitation is a process for obtaining and synthesizing the judgments of subject-matter experts when the available knowledge base (e.g., empirical data and formal models) is incomplete, unreliable, uncertain, or open to alternative interpretation. Expert elicitation can be used to

- Predict future events;
- Provide estimates on new, rare, complex, or poorly understood phenomena;
- Integrate or interpret existing information; or
- Determine what is currently known, how well it is known, or what further exploration is needed about a subject or field.

Expert elicitation has been applied in a wide range of fields including probabilistic seismic hazard analysis (Budnitz et al., 1998), analysis of the health impacts of air pollutants (Cooke et al., 2007), assessment of the impact of new train technologies on human reliability (Wreathall et al., 2004), and analysis of risks of terrorism and the effectiveness of security policies to reduce those risks (DHS Bioterrorism Risk Assessment as reported by Ezell et al., 2010).

Expert elicitation can be used in risk assessment to obtain expert judgments about the three elements of the risk triplet:

- What can go wrong? (scenarios)
- How likely is that to happen? (likelihoods)
- What are the consequences if it does happen? (consequences)

Expert elicitation can also be informal or formal. Informal methods for eliciting expert judgements, although sometimes producing good results, are usually problematic because they have no built-in controls for bias, relevance, and consistency of knowledge across experts or variability in interpreting the questions posed. The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum (GIF, 2011) notes, "Without a formal process and strong controls, experts may be asked to provide judgments on issues that go beyond their expertise, or their estimates might be combined in misleading ways which distort the results" (p. 67).

More formal structured expert elicitation methods have been developed to overcome the limitations of informal methods (Budnitz et al., 1998; Keeney and von Winterfeldt, 1991; Morgan, 2014). Formal expert elicitation is a structured process that makes use of people who are sufficiently knowledgeable about particular subject areas to make meaningful assessments. Key elements of formal expert elicitation processes include

- Careful selection of experts to ensure broad representation of relevant areas of expertise and perspectives;
- Training for elicitation, including sensitization to known cognitive biases;
- Providing models and tools to support problem formulation and exploration;
- Providing opportunity for extensive, highly structured expert interaction to maximize a shared understanding of the available relevant empirical database, models, and reasoning processes; and
- Uncovering and documenting areas of clear agreement as well as legitimate diversity of assessment.

The use of formal processes for expert elicitation has improved the credibility and acceptance of expert judgment because of the rigor and transparency of the results (Budnitz et al., 1998).

A good example of an expert elicitation application is provided by Budnitz et al. (1998). They developed and exercised methodological guidance on how to perform a probabilistic seismic hazard analysis that relied heavily on expert elicitation. Expert elicitation was needed in this case because there were major limitations in the research community's understanding of the mechanisms that cause earthquakes and the processes that govern how an earthquake's energy propagates, despite advances in seismic knowledge. The authors leveraged the knowledge of experts in seismic analysis to develop estimates of the likelihood of various levels of earthquake-caused ground motions at particular locations for a given future time period that reflected the current state of knowledge.

Among the major methodological contributions the authors made to the expert elicitation literature was to explicate the various types of consensus that can exist in a group of experts. For example, experts may all agree on the same deterministic model or the same value for a particular variable. Alternatively, experts may differ on particular models or parameter values but agree that a particular composite probability distribution represents the composite beliefs of the overall scientific community.

Budnitz et al. (1998) argued that it is far easier to get a group of experts to agree on how to represent the informed community's diversity of beliefs about a scientific issue than it is to get them to agree on the resolution of a particular technical issue. As a consequence, the probability distributions produced using formal expert elicitation can produce an accurate representation of the level of uncertainty associated with particular likelihood assessments.

The successful application of expert elicitation methods to security risk assessments depends on the rigor and transparency of the methodologies employed. Of particular importance is the selection of experts who collectively possess the necessary range of expertise needed to develop scenarios, event/fault trees, likelihood estimates, and uncertainty ranges. This would include individuals from the intelligence community with access to knowledge about terrorist motivations, intent, and capabilities. It would also include individuals from the security community who understand capabilities to prevent or respond to attacks, as well as experts in physical systems to assess the physical consequences of (low-likelihood) successful attacks.

Another key element of expert elicitation is to provide models and tools to support problem formulation and exploration. In the case of terrorist threat this includes providing models and tools that encourage consideration of terrorists as intelligent, goal-driven adversaries that gather information about our own defensive preparations and seek to maximize the achievement of their own objectives. The CARVER analysis method described in Appendix 5B is one example of a tool that encourages domain experts to consider the attack space from the perspective of an intelligent, motivated adversary. Decision trees, attack trees, and game-theoretic formulations are other examples of models and tools that can provide structure for eliciting, representing, and exploring the consequences of interaction among multiple intelligent agents that include adversaries and defenders. Game-playing exercises (e.g., red teams and cyber hacking teams) may also be useful.

The USNRC has considerable experience in the use of expert elicitation and has issued guidance documents on its use (Kotra et al., 1996; see also Budnitz et al., 1998; Forester et al., 2007). The USNRC can draw on this experience to develop security risk assessments for which limited observational data are available (Frye, 2013). Additionally, there continues to be active research on methods for eliciting and combining expert assessments to improve the sharpness[1] and reliability of estimates that can also be drawn upon (Mellers et al., 2015; Satopää et al., 2014; Wang and Bier, 2012).

[1] The *sharpness* of an analysis describes the ability to differentiate among its results or outcomes. The greater the sharpness of the analysis, the greater the confidence that can be placed on the uncertainty ranges of its outcomes. See Satopää et al. (2014) and Gneiting et al. (2007).

APPENDIX 5B
CARVER Analysis

The WWII U.S. Office of Strategic Services developed a targeting doctrine for optimizing scarce resources in attacks on the German military in then-occupied Europe. This targeting methodology has been used by U.S. Special Operations for decades to plan for small unit raids and has been employed by the DHS and numerous international private security enterprises. It can also be used as a qualitative vulnerability analysis tool. This methodology is referred to as the CARVER Vulnerability Assessment Methodology. This methodology has been used by the U.S. government for more than 40 years (USNRC, 2007a, Enclosure 5; see also Bennett, 2007).

CARVER is defined as follows:

CARVER = Criticality + Accessibility + Recuperability + Vulnerability + Effect + Recognizability[1]

The CARVER factors are described in the Department of the Army Field Manual (AFM, 1991). The following descriptions of a CARVER analysis are taken from that manual:

- *Criticality:* A target (or target-specific critical node) is considered critical when its partial or complete destruction has significant military, political, psychological, or economic operational impacts. Evaluation of critical nodes or single points of failure associated with a given target is done within the context of the target's primary mission.
- *Accessibility:* A target is considered accessible when sufficient personnel and equipment can physically emplace explosives or other devices or employ stand-off weapons to degrade or destroy it. The evaluation of accessibility requires the identification of critical operational paths to achieve the mission objectives and factors that can aid or impede target access.
- *Recuperability:* A target's recuperability is the length of time and level of effort required to repair, replace, or bypass damage or destruction and restore mission capability.
- *Vulnerability:* A target is vulnerable if the attacking force has the means and the expertise to achieve the desired effect. When determining the vulnerability of a target, the ability to disrupt or

[1] George C. Marshall Center for Security Studies, The Executive Course, Distance Learning Seminar #732, Terrorism and Its Implications for Democratic States (http://pubs.marshallcenter.org/732/lesson1/lesson1.html).

destroy a critical component is compared with the attacking force's operational capabilities and weaponry.

- *Effect:* An attack on a target may have desirable as well as undesirable military, political, economic, psychological, and sociological impacts. Effect is a measure of these impacts. Planners can use this factor in conjunction with the criticality factor to select particular targets for attack. From a terrorist's perspective, special consideration is given to the effect on local populations, potential for media interest, potential psychological and sociological impacts, and possible effects on the target nation's political and economic systems.
- *Recognizability:* The target is recognizable if it can be identified by the attacker(s), intelligence collection team, or reconnaissance element under varying conditions and circumstances.

The CARVER selection factors assist in determining the best targets or components of targets to attack. Each of the factors above is given a numerical value for the targets being considered. This value represents the desirability of attacking the target. The values are then placed into a decision matrix and summed for each target. The targets with the highest values are considered to be the best targets to attack, giving consideration to existing and future operational constraints.

6

Loss-of-Coolant Events in Spent Fuel Pools

This chapter provides supporting information for the present committee's findings and recommendations in Chapter 4 (Table 4.1) related to understanding and mitigating loss-of-coolant events in spent fuel pools:

FINDING 4.5: Technical analyses undertaken by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories after 2004 confirm that reconfiguring spent fuel in pools can be an effective strategy for reducing the likelihood of fuel damage and zirconium cladding fires following loss-of-pool-coolant events. However, reconfiguring spent fuel in pools does not eliminate the risks of zirconium cladding fires, particularly during certain periods following reactor shutdowns or for certain types of pool drainage conditions. These technical studies also illustrate the importance of maintaining water coolant levels in spent fuel pools so that fuel assemblies do not become uncovered.

FINDING 4.6: Additional analyses and physical experiments carried out by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories since NRC (2006) was completed have substantially improved the state of knowledge of boiling water reactor (BWR) and pressurized water reactor (PWR) spent fuel behavior following partial or complete loss of pool water. These studies and experiments have addressed the following important issues:

- **Fuel damage state and timing as a function of fuel age and pool water loss.**

- Propagation of zirconium cladding fires to other assemblies in the pool.
- Potential mitigation strategies (dispersion of hot fuel assemblies in the pool, water sprays, water replacement) for delaying or preventing fuel damage following pool water loss.

These experiments have resulted in significant validation of the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code that is used to model coolant loss in spent fuel pools. However, the code is unable to adequately model flows when stratification occurs and plumes form in the pool and/or above-pool environment. Moreover, key portions of code lack validation, and there has been no end-to-end validation of the code for modeling coolant loss in spent fuel pools.

RECOMMENDATION 4.6 The U.S. Nuclear Regulatory Commission should (1) sponsor an end-to-end validation of the MELCOR code for use in modeling coolant loss in spent fuel pools, and (2) validate key submodels in the code with particular attention paid to

- **Modeling the thermal and chemical behavior of spent fuel assemblies in partially drained pools.**
- **Modeling the thermal and chemical response of spent fuel assemblies to the application of water sprays.**
- **Modeling and validating for stratified flows in fully and partially drained pools.**

FINDING 4.8: The U.S. Nuclear Regulatory Commission and the U.S. nuclear industry have made good progress in implementing actions to address Recommendation 3E-2 in NRC (2006). The U.S. Nuclear Regulatory Commission has directed plant licensees to

- **Reconfigure their spent fuel in pools to achieve at least a 1 × 4 dispersion of high- and low-decay-heat assemblies, unless such configuration can be shown to be inapplicable or unachievable. This configuration must be achieved following each fuel offload from the reactor not later than 60 days after reactor shutdown.**
- **Develop guidance and implement strategies to maintain and restore spent fuel pool cooling following explosions and fires. To address this requirement, the U.S. nuclear industry has developed and adopted guidance and strategies for spent fuel pool water makeup and water sprays.**

However, additional work is needed to more fully implement Recommendation 3E-2 in NRC (2006).

RECOMMENDATION 4.8: The U.S. Nuclear Regulatory Commission should take the following actions to more fully implement Recommendation 3E-2 in NRC (2006):

- **Reexamine the need for the 60-day limit for fuel dispersion and reduce the allowable time if feasible.**
- **Reexamine and, if needed, redesign the water makeup and spray systems and strategies to ensure that they can be implemented when physical access to pools is hindered or the site becomes inaccessible.**

This chapter is organized as follows:

- Section 6.1 provides supporting information for **Findings 4.5** and **4.6** and **Recommendation 4.6** on understanding the response of spent fuel pools to loss-of-coolant events.
- Section 6.2 provides supporting information for the present committee's **Finding 4.8** and **Recommendation 4.8** on mitigating loss-of-coolant events in spent fuel pools.

6.1 RESPONSE OF SPENT FUEL POOLS TO LOSS-OF-COOLANT EVENTS

The U.S. Nuclear Regulatory Commission (USNRC) and its contractor Sandia National Laboratories (Sandia) have expended considerable effort to understand the response of spent fuel pools to loss-of-coolant events since NRC (2006) was published:

- Physical experiments have been conducted on full-sized BWR and PWR assemblies to study fuel cladding ballooning and rapid zirconium oxidation (Lindgren and Durbin, 2007; NEA, 2015).
- Various loss-of-coolant scenarios have been examined for BWR and PWR assemblies in pools using an improved version of the MELCOR code (see Sidebar 6.1) that addressed some of the deficiencies identified in NRC (2006). Many of the publications from these studies contain security-related information and have not been released to the public. These publications are described in Appendix 6A of this chapter.

The committee finds (**Finding 4.6**; see Table 4.1 in Chapter 4) that these analyses and physical experiments have substantially improved the

SIDEBAR 6.1
Modeling Loss-of-Pool-Coolant Events

At present, there is no single modeling approach or software tool that can be used to model all aspects of loss-of-coolant accidents in spent fuel pools. Both integral models (MELCOR) and computational fluid dynamics (CFD) models have been used to examine spent fuel pool behavior in water-filled, partially drained, and fully drained pools. There are advantages and disadvantages to both types of models.

MELCOR

MELCOR is a lumped-parameter control volume model that was originally developed to model severe accidents in reactor cores, including uncovering of the reactor core, fuel damage, hydrogen generation, and release of radioactive material from a variety of accident initiators including unrecovered LOCA, station blackout, and other unrecovered accidents. As a general purpose tool, it is useful for modeling not only core degradation but also the subsequent movement of molten core material outside the reactor pressure vessel, generation and transport of combustible gases, and release and transport of fission products through the containment and the reactor building. Ross et al. (2014) provide an extensive discussion of the validation and independent peer review of MELCOR as applied to the State-of-the-Art Reactor Consequence Analyses (SOARCA; USNRC, 2012b). See especially Table 4.2 of Ross et al. (2014) for a historical summary of validation for both in-vessel and ex-vessel accident phenomena. Spent fuel pool accidents were not included in the SOARCA.

MELCOR also has been used to model loss-of-coolant accidents in spent fuel pools. This application makes use of MELCOR capabilities to model heat transfer, chemical reactions, fluid mechanics, and fission product release. The pool and fuel are divided into a set of control volumes, which correspond to an axial subdivision of each fuel assembly, and large volumes representing the open regions within the pool and the enclosure above the pool. This approach is valid for modeling flow around fuel assemblies in close-racked configurations as long as the racks and fuel assemblies are intact. MELCOR has not been applied to modeling flow around fuel assemblies in open-rack configurations. Validating experiments or CFD simulations would be a necessary activity in order to have confidence in the results.

The code can treat radial flow in open racks as long as crossflow resistances and flow areas are provided. However, no validation of calculated crossflows exists.

There are also models within MELCOR for dealing with the late stages of a loss-of-pool-coolant accident such as the generation of a debris bed and interaction of molten fuel with the pool liner and concrete. However, the outcome of an accident under these situations is more uncertain than in early stages when fuel geometry is intact. Moreover, even during the early stages, there are limitations in using lumped-parameter control volumes in the open regions away from the fuel racks and assemblies.

continued

SIDEBAR 6.1 Continued

The physical domain in the model is broken up into a set of volumes and associated averaged or lumped values for physical properties such as density, temperature, and pressure for each material (air, steam, fuel, and cladding) within the volume. These lumped control volumes are connected by a set of flow paths (junctions) according to a user-defined scheme, and simple flow-resistance models are used to compute the junction flow between control volumes based on pressure differences. Submodels are provided for heat transfer to the pool wall and other masses such as racks. One-dimensional steady-state correlations are used to treat pressure drops within the channels defined by the junction flow within channel boxes and rack partitions.

From a geometric point of view, MELCOR is able to model control volumes with multiple pathways in transverse directions. From a fluid dynamics point of view, MELCOR does not have a bona fide momentum equation; rather, it models pressure-driven flows through flow junctions between control volumes. Flow rate is controlled by friction factors and flow area and pressure drop correlations (see Chapter 2 of the MELCOR user manual[a]).

The lack of a momentum equation limits the usefulness of MELCOR for modeling multidimensional fluid mechanics. Momentum effects are important for calculation of two- or three-dimensional flows. MELCOR can model buoyancy-driven flows within fuel racks and fuel assemblies, but it is one dimensional.

Some of the recent work at Sandia National Laboratories validates some portions of the MELCOR application to spent fuel pools, but validation does not appear to have been carried out for aspects involving natural convection, coupled pool-fuel assembly behavior, or spray cooling. In the absence of three-dimensional calculations, it is not possible to quantify how the omission of a momentum equation and multidimensional buoyancy-driven flow affects model results for a variety of spent fuel pool scenarios.

[a] Available at http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6119/v2/.

state of knowledge of BWR and PWR spent fuel behavior following partial or complete loss of pool water. These analyses and experiments addressed three of the questions raised in Recommendation 3E-1 (NRC, 2006; see Table 4.1 in Chapter 4):

- Is it feasible to reconfigure the spent fuel within pools to prevent zirconium cladding fires[1] given the actual characteristics (i.e., heat

[1] As noted in Chapter 4, the term *zirconium cladding fire* is used to describe the self-sustaining oxidation of zirconium fuel cladding. This oxidation results in a temperature runaway that can generate enough heat to melt the fuel pellets.

CFD MODELING

Accurate modeling of natural convection in large open volumes in a spent fuel pool, including three-dimensional flows of air through spent fuel racks (particularly when open racks are installed in a pool), requires the solution of field equations for mass, momentum, and energy conservation on a two- or three-dimensional grid or spatial network. This is possible with CFD modeling. However, because of limitations on execution time and computer storage (even with modern supercomputers), model grids are too coarse to represent all physical processes. This means that appropriate subgrid-scale models for interphase mass, momentum and energy exchange, turbulent flow processes, and chemical reaction must be used to close the model.

It is impractical to model fluid mechanics with this fidelity while modeling radiative and two-phase (nucleate and film boiling) heat transfer at hot surfaces, oxidation of the fuel cladding (as well as grids and channel boxes), phase changes of materials (zirconium cladding, steel structures, and uranium oxide fuel), as well as fission product release—all essential processes that need to be incorporated into severe accident modeling. This difficulty is at the core of severe accident modeling. There are many generations of efforts to develop CFD approaches to severe accident modeling; at present, they remain restricted to particular aspects of accidents such as thermal hydraulics, hydrogen mixing or combustion, and fuel-coolant interactions rather than integral approaches as is possible with MELCOR.

Separate simulations have to be carried out with CFD models to accurately represent the fluid dynamics of multidimensional buoyancy-driven flow for natural convection in open volumes or large enclosures. It is not possible to include realistic models of fuel assemblies because of storage and execution time limitations. Instead, a "porosity" model is used to represent the internal structure of the fuel assemblies. Flow paths inside the racks and fuel assemblies are characterized using a flow resistance model, very similar to the approach in MELCOR. Chemical reactions and heat transfer between the water or steam and fuel are typically not included.

See Sehgal (2011) for a discussion of the value and motivations for using integral codes such as MELCOR and the advantages and disadvantages of multidimensional approaches such as CFD.

generation) of spent fuel assemblies in the pool, even if the fuel were damaged in an attack?

- In the event of a localized zirconium cladding fire, will such rearrangement prevent its spread to the rest of the pool?
- How much spray cooling is needed to prevent zirconium cladding fires and prevent propagation of such fires?

The committee finds (**Finding 4.5;** see Table 4.1 in Chapter 4) that technical analyses undertaken by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories confirm that reconfiguring spent fuel in pools can be an effective strategy for reducing the likelihood of fuel damage and zirconium cladding fires following loss-of-pool-coolant events. Indeed, the analyses indicate that fuel configuration in a pool can affect its coolability[2] in air following a loss-of-pool-coolant event. There are certain pool water states and time periods when the fuel is not coolable in air and is therefore subject to damage (e.g., fuel rod ballooning) and rapid oxidation (i.e., zirconium cladding fires), regardless of its configuration in the pool.

The following sections provide brief summaries of this USNRC and Sandia work. Some details of the analyses, for example, the particular conditions under which fuel damage and zirconium cladding fires can occur, as well as the timing of such occurrences, are not provided in this report because they are security sensitive.

6.1.1 Physical Experiments

In the late 1990s, USNRC staff evaluated spent fuel pool accident risks at decommissioned nuclear plants in the United States. The results of this evaluation are documented in Collins and Hubbard (2001). Some assumptions in the accident progression were known to be conservative, especially fuel damage estimates. The USNRC subsequently initiated efforts to evaluate severe accident progression in spent fuel pools at operating plants using best-estimate computer codes. These code calculations identified various modeling and phenomenological uncertainties.

Following the release of the NRC (2004) report, the USNRC initiated an experimental program at Sandia to investigate thermal-hydraulic phenomena associated with complete loss-of-coolant accidents in spent fuel pools in light-water reactors. The ultimate objective of this program was to simulate accident conditions of interest for the spent fuel pool in a full-scale prototypic fashion (electrically heated, prototypic assemblies in a prototypic spent fuel pool rack). A major impetus for this work was to allow code validation (primarily MELCOR) and reduce modeling uncertainties. The results of this work are documented by Lindgren and Durbin (2007).

As shown in Table 6.1, Sandia used a phased approach with three basic types of experiments to complete this program. As a proof of concept, two heater-design tests were first performed to determine the suitability of the electrically heated, zirconium-clad spent fuel rod simulators. Three separate-effects tests were then conducted to study specific phenomena

[2] The committee uses the term *coolability* to indicate whether air cooling of stored fuel is sufficient to prevent self-sustaining (runaway) oxidation of its zirconium cladding. Runaway oxidation may occur in the range 900°C-1200°C depending on thermal-hydraulic conditions. Fuel that is coolable in air will not reach these temperatures. Section 6.1.3 in this chapter describes the reactions that occur if fuel reaches this temperature.

TABLE 6.1 Description of Tests in Lindgren and Durbin (2007)

Test Description	Purpose	Assembly	Rod Material
Heater Design	Test electrical heater performance, preliminary data on zirconium fire; conducted at normal and reduced oxygen concentrations	12-rod bundle	Zircaloy
Separate Effects	Hydraulics: Determine viscous and form loss coefficients for laminar volumetric flow rates	Prototypic	Stainless steel
	Thermal hydraulics: Determine input conditions for partial-length experiments	Prototypic	Incoloy
	Thermal radiation: Determine radiation coupling in a 1 × 4 arrangement	Prototypic: Partial length	Incoloy
Integral Effects	Axial ignition: Determine temperature profiles, induced flow, axial O_2 profile, nature of rapid zirconium oxidation	Prototypic: Single full-length assembly	Zircaloy
	Determine radial fire propagation in a 1 × 4 arrangement	Partial-length assembly	Zircaloy

such as heat transfer (e.g., thermal radiative coupling) and fluid flow (e.g., induced natural convective flow). These tests were nondestructive and involved some nonprototypic materials (e.g., stainless steel and Incoloy). Finally, two prototypic assemblies were heated to ignition in a series of integral-effects tests.

The heater-design tests were conducted with a 12 × 12 rod bundle configuration with zirconium cladding. Sandia researchers demonstrated that ignition was possible when the test design minimized heat loss and maximized gas preheating and bundle power.

The separate-effects tests utilized a single full-length or partial-length "highly prototypic" BWR 9 × 9 fuel rod assembly. Sandia researchers measured the thermal-hydraulic response and determined appropriate loss coefficients as a function of bundle mass flow under adiabatic conditions.

The integral-effects tests used five one-third-length zirconium fuel assemblies arranged in a 1 × 4 configuration (i.e., a center assembly and four face assemblies; see Figure 7.2 in Chapter 7) in a 3 × 3 pool rack. The tests were designed to simulate the middle to upper portions of an array of full-length assemblies.

An important aspect of this project was the deliberate close coupling of experiments with numerical analysis. The project utilized the severe accident code MELCOR as (1) a tool for the experimental design, (2) for pretest results prediction, and (3) for post-test analysis of the calculated and measured responses. The post-test MELCOR analysis helped refine some model parameters, which led to improvements in the fidelity of model predictions.

6.1.2 MELCOR Analyses

The MELCOR code (Sidebar 6.1) was originally developed for analysis of severe accidents in BWR and PWR reactors. The code is based on control volumes connected by flow paths that are treated as "pipes" with specified frictional losses. Flow areas are either given or are determined as degraded material relocates. Generally, one-dimensional (1D) flows are considered. However, the code can be exercised in a pseudo-2D manner to calculate radial flow rates.

MELCOR has been updated for use in investigating conditions in spent fuel pools under various loss-of-coolant conditions. The updated version of MELCOR includes models for

- Fuel degradation;
- Radiative, convective, and conductive heat transfer;
- Air and steam oxidation;
- Hydrogen production and combustion;
- Boiling and two-phase thermal hydraulics; and
- Fission-product release and transport.

Recent enhancements to the code include a new air-oxidation kinetics model and a new spent fuel assembly flow-resistance model based on the experiments described in the previous section of this chapter. Nevertheless, the MELCOR model still has several limitations:

- It cannot model stratified flow or buoyancy-driven flow (i.e., formation of plumes and circulatory flow patterns in large spent fuel pools or above-pool environments) or open-rack configurations in which fuel assemblies are not contained in solid-wall boxes as is current practice in dense-packed pools.
- It cannot model two-phase flow and boiling heat transfer when structural debris falls into the fuel assemblies in the pool.
- It cannot model nitriding reactions with zirconium.
- For spray-cooling scenarios (i.e., when water is sprayed onto the tops of the fuel assemblies in the pool to cool them), it cannot model entrainment and deentrainment of water droplets.[3] Droplet

[3] Wallis' correlation is used for the flooding limit, but spray penetration depth calculations are suspect.

behavior during travel from the spray nozzle to the fuel assemblies is not modeled, and modeling of heat transfer from the cladding to the impacting droplets is lacking.

- It cannot model the simultaneous flow of air, steam, or water droplets through fuel assemblies during spray cooling of a fully drained pool.
- Fuel cladding degradation models are empirical and are based on user-specified criteria.

MELCOR has been used by Sandia to analyze complete and partial loss-of-coolant scenarios at the fuel assembly and whole-pool level. These analyses show that several factors affect spent fuel coolability, including

- Fuel aging time,
- Fuel configuration in the pool,
- Size and location of coolant leaks in the pool,
- Ventilation above the pool,
- Radial thermal coupling, and
- Deformation of the fuel rod bundle geometry.

These factors are described below.

6.1.2.1 Fuel Aging Time

Fuel age refers to the elapsed time since the fuel was in an operating reactor. Once the reactor is shut down, heat production from radioactive decay in the fuel decreases rapidly. In general, the older the fuel, the less heat it generates. If the fuel is offloaded from the reactor to the pool, there is a certain period of time after offload during which the fuel is not coolable in air, even if the pool is completely drained (see, for example, USNRC [2013], which is discussed in Chapter 7).

6.1.2.2 Fuel Configuration in the Pool

Sandia analysts used MELCOR to examine five configurations of fuel storage in spent fuel pools:

1. A uniform configuration in which all of the fuel assemblies from a reactor offload are grouped together in the pool;
2. A 1 × 4 configuration in which a hotter fuel assembly is surrounded on four sides with colder assemblies;
3. A 1 × 4 configuration in which some of the surrounding cold assemblies are missing (i.e., the spent fuel rack is empty in some locations);

4. A checkerboard configuration of hot and cold assemblies; and
5. A checkerboard configuration where some of the surrounding cold assemblies are missing.

The 1 × 4 and checkerboard configurations are illustrated in Figure 7.2 in Chapter 7. When higher-decay-heat (i.e., hotter) assemblies are surrounded by lower-decay-heat (i.e., colder) assemblies, the temperature rise of the hotter assemblies is slowed, mostly because of heat loss by radiation to the colder assemblies and their thermal inertia. In other words, the thermal capacities of the colder assemblies play an important role in regulating temperature rise of the hotter assemblies.

The Sandia analyses showed that, at higher temperatures, the heat-up rate of 1 × 4 configurations is slower compared to the other configurations studied. They also showed that zirconium cladding fires, once initiated, can spread to other assemblies in the pool. At high temperatures, fuel rod integrity is lost and fuel material is rearranged into debris piles or molten pools. Radioactive materials are released from the fuel as noble gases and aerosols (see Sidebar 2.2 in Chapter 2).

6.1.2.3 Size and Location of Coolant Leaks in the Pool

Development of an uncontrolled leak on the pool boundary can lead to the loss of pool coolant. The location of the leak determines whether the pool will drain partially or completely. The size and location of the leak will determine how long it will take to drain the pool.

Sandia analysts used MELCOR to investigate the effect of leak size and location on air coolability limits of BWR and PWR assemblies stored in spent fuel pools. Two scenarios were considered:

- Complete loss of coolant, where the pool was assumed to drain below the base plate of the spent fuel racks, allowing for natural convection of air through the assemblies, and
- Partial loss of coolant, where the pool was not drained completely, so water covered the lower portions of the fuel assemblies. This blocked airflow through the assemblies until water in the lower portion of the pool boiled off.

For both scenarios, it was assumed that spent fuel cooling and building ventilation systems were not functioning and that no water was being added to the pool.

6.1.2.4 Ventilation Above the Pool

The Sandia analyses showed that ventilation of the enclosed space above a spent fuel pool affects the coolability of stored fuel in air. Under poor ventilation conditions (e.g., when the building ventilation system is not functioning) in a fully drained pool, stored fuel of a certain age[4] can undergo self-sustaining oxidation (zirconium cladding fire) that can spread to other assemblies in the pool and release fission products to the environment. However, fission-product aerosols are mostly retained in the pool enclosure as long as it is not damaged. If the enclosure is damaged, for example by a hydrogen explosion, then fewer aerosols will be retained. Under good ventilation conditions (e.g., the blowout panels above the spent fuel pool are open), the fuel may be coolable in air under complete loss-of-coolant conditions, again depending on its age, but the fuel may not be coolable in air under partial-loss-of-coolant conditions.

6.1.2.5 Radial Thermal Coupling

Heat transfer from hot fuel assemblies to adjacent cold assemblies in a spent fuel pool plays an important role in the propagation of zirconium fires and the consequent release of radioactive material from the fuel. Radiation is the dominant mode of heat transfer at high temperatures, so accuracy in the calculation of radiative heat transfer is important for estimating assembly temperatures. The limiting condition occurs when a hot assembly is totally isolated from surrounding colder assemblies. In this case, the heat-up rate of the hot assembly will be much higher, leading to early fuel damage and radioactive material releases. In the absence of heat gain from the hot assemblies, the heat-up rate of the colder fuel assemblies will be lower, limiting the spread of a zirconium cladding fire from hotter to colder assemblies. MELCOR sensitivity analyses of fuel heat-up have been carried out for three different radial thermal coupling configurations (i.e., no coupling, normal coupling, and reduced coupling).

6.1.2.6 Deformation of the Fuel Rod Bundle Geometry

Internal pressures in the fuel rods can cause localized ballooning as mechanical properties of the cladding degrade with increases in temperature. The occurrence of ballooning can increase flow resistance within the fuel assembly and impair cooling, possibly leading to further cladding degradation. Clad ballooning cannot be modeled mechanistically using

[4] The exact ages of the fuel for this condition is security-related information and therefore not disclosed in this report.

MELCOR but is instead modeled parametrically by reducing the flow area. A co-planar blockage is created at the exit of the rod bundle when the peak temperature exceeds an established criterion for the onset of ballooning. A more than minor reduction in flow area,[5] which occurs when neighboring fuel rods in an assembly are in physical contact, will produce a small increase in the calculated maximum cladding temperature but will have little effect on the initiation of cladding oxidation.

Crushing of fuel rods due to mechanical loading can also reduce flow areas and impose additional flow resistance. MELCOR was used to study the effects of crushing on the coolability of spent fuel in air for a completely drained pool. Crushing was assumed to occur along the entire length of the fuel rod, and flow area was reduced parametrically. The reduced flow area allows temperatures in the cladding to increase sufficiently to cause self-sustaining oxidation, even in older spent fuel. However, when flow-area reductions are high,[6] the reduced rate of air flow through the assemblies limits the extent of oxidation and thus slows the rise of cladding temperatures, delaying the onset of self-sustaining oxidation conditions.

A reduction in flow area and accompanied increase in the flow resistance can also occur if structural debris falls onto the spent fuel (such as occurred at the Fukushima Daiichi plant; see Chapter 2). The flow resistance and thermal insulation created by this debris can reduce the coolability of the stored fuel and accelerate fuel rod heat-up. To the committee's knowledge, this scenario has not been evaluated by the USNRC or Sandia.

Sandia has carried out hand calculations to assess the coolability of debris beds formed from the relocation of degraded fuel rods, cladding, and structural material to the bottom of a spent fuel pool. Both liquid-saturated debris beds with an overlying layer of liquid and dry debris beds have been considered. The coolability limit strongly depends on effective particle diameter, porosity, and depth of the debris bed. Calculations have also been carried out for a dried-out debris bed that is cooled by the flow of air. The coolability of the fuel in the debris bed depends on the bed's effective particle diameter and porosity.

6.1.3 Spent Fuel Pool Loss-of-Coolant Accidents (LOCAs)

In a complete-loss-of-pool-coolant scenario, most of the oxidation of zirconium cladding occurs in an air environment:

$$Zr + O_2 \rightarrow ZrO_2.$$

[5] The specific percentage is security-related information.

[6] The specific percentage is security-related information.

For a partial-loss-of-pool-coolant scenario (or slow drainage in a complete-loss-of-pool-coolant scenario), the initial oxidation of zirconium cladding will occur in a steam environment:

$$Zr + 2H_2O_{steam} \rightarrow ZrO_2 + 2H_2.$$

Both of these reactions are highly exothermic. The zirconium-steam reaction leads to the formation of hydrogen, which can undergo rapid deflagration in the pool enclosure, resulting in overpressures and structural damage. This damage can provide a pathway for air ingress to the pool, which can promote further zirconium oxidation and allow radioactive materials to be released into the environment. Debris from the damaged enclosure can fall into the pool and block coolant passages.

The MELCOR analyses show that fuel in a completely drained pool can be more easily cooled by air than in a partially drained pool. Coolability in a completely drained pool is promoted by the establishment of natural air circulation through the fuel assemblies.

In a partially drained pool, cooling of the uncovered portions of fuel assemblies occurs mainly by steam that is generated as the pool water boils off. The steam production rate depends on the decay-heat generation rate in the fuel and the portion of the fuel rods that are covered with the two-phase mixture of steam and water. The rate affects the temperature rise in the dry regions of the rods and the rate of hydrogen production as a result of the zirconium-steam reaction. The exothermic zirconium-steam reaction also adds heat to the fuel rod.

After the water level drops below the rack base plate, convective air flow is established. If the steam is exhausted, then the zirconium-steam reaction is replaced by the zirconium-oxygen reaction. However, prior to the onset of convective air flow, fuel cladding temperatures can exceed the threshold for oxidation, and fuel damage and radioactive material release can occur. The time to damage and release depends on pool water depth relative to the stored fuel assemblies.

Significant validation of the MELCOR code and improvement of thermal, chemical, and mechanical models has been carried out with experiments on electrically heated rods to examine the behavior of spent fuel assembly heat-up, oxidation, and loss of coolable geometry in air environments corresponding to a completely drained pool. However, no experimental validation of these phenomena has been performed for a partially drained pool. The present committee's **Recommendation 4.6** (see Table 4.1 in Chapter 4) calls for validation of the thermal and chemical behavior of spent fuel assemblies in partially drained pools. This validation is important because, as noted previously, there is a higher hazard for zirconium cladding fires in partially drained pools.

6.1.4 Mitigation Strategies for Spent Fuel Pool LOCA

At least two mitigation strategies are available to mitigate a loss-of-coolant event in a spent fuel pool: Repair the leak that is causing water to be lost and/or add makeup water. In the absence of leak repair, the location, size of the leak, magnitude of decay heat in the pool, and rate and timing of makeup water addition determines the effectiveness of the mitigation strategy. Sandia used MELCOR to determine the desired flow rate of makeup water under different accident scenarios. In the case where a spent fuel pool drains completely, adding makeup water will cover the lower portions of the fuel assemblies and block air convection. This could lead to heat-up of the fuel and production of hydrogen as a result of zirconium-steam reaction, loss of coolable geometry in the assembly, and eventually self-sustaining oxidation of zirconium. The flow of makeup water must be high enough to cover a certain portion[7] of the active fuel height before these conditions occur.

Spraying water on top of the fuel assemblies may also be an effective strategy to provide additional cooling if makeup water capabilities are inadequate to maintain pool water levels above the tops of the fuel racks. Sandia used MELCOR to study the effectiveness of spraying with a certain flow rate and delay time after a loss-of-pool-coolant event. Analysts found that, for certain fuel configurations,[8] spraying the fuel can be an effective strategy for maintaining coolability of the fuel. To the committee's knowledge, no experimental verification of MELCOR calculations with respect to droplet size, effect of counterflowing steam and/or gases on droplet carryover, and wetting of surface by droplets and associated heat transfer has been provided.

The present committee's **Recommendation 4.6** (see Table 4.1 in Chapter 4) calls for the validation of the thermal and chemical response of spent fuel assemblies to the application of water sprays. Analysis should be carried to determine the envelope of water flow-rate conditions, either as water makeup or spray, for fuel assembly coolability. The presence of debris in the pool, which can block water sprays, should be considered in the assessment.

Validating the responses of stored fuel to water sprays is essential for confirming the effectiveness of existing mitigation capabilities for loss-of-coolant events in spent fuel pools. These capabilities are discussed in the next section.

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[7] The exact height is security-related information and is not disclosed in this report.

[8] The exact configuration is security-related information and is not disclosed in this report.

6.2 MITIGATING LOSS-OF-COOLANT EVENTS IN SPENT FUEL POOLS

NRC (2006) recommended that the USNRC ensure that nuclear plant operators take prompt and effective measures to reduce the consequences of loss-of-pool-coolant events that could result in propagating zirconium cladding fires. Two specific measures were recommended for prompt implementation (Recommendation 3E-2 [NRC, 2006]; see Table 4.1 in Chapter 4):

1. Reconfigure the fuel in the pools so that high-decay-heat fuel assemblies are surrounded by low-decay-heat assemblies, and
2. Make provision for water-spray systems that would be able to cool the fuel even if the pool or overlying building were severely damaged.

The present committee finds (**Finding 4.8**; see Table 4.1 in Chapter 4) that the USNRC and the U.S. nuclear industry have made good progress in implementing this NRC recommendation. The USNRC has directed plant licensees to

- Reconfigure their spent fuel in pools to achieve at least a 1 × 4 dispersion of high- and low-decay-heat assemblies (see Figure 7.2 in Chapter 7), unless such configuration can be shown to be inapplicable or unachievable. This configuration must be achieved following each fuel offload from the reactor not later than 60 days after reactor shutdown.
- Develop guidance and implement strategies to maintain and restore spent fuel pool cooling following explosions and fires.

The USNRC informed the committee that most U.S. nuclear plants have implemented the first recommended action. Additionally, the U.S. nuclear industry has developed and adopted guidance and strategies to implement the second recommended action. Further discussion is provided below.

Following the September 11, 2001, terrorist attacks on the United States, the USNRC issued Order EA-02-026 (Order for Interim Safeguards and Security Compensatory Measures).[9] Section B.5.b of the order directed nuclear plant licensees to develop and implement strategies to maintain or restore core, containment, and spent fuel pool cooling capabilities following large explosions or fires that damaged large areas of the plant. The order

[9] The order is designated as Safeguards Information and has not been released to the public, but its requirements have been codified in 10 CFR 50.54(hh)(2).

directed licensees to identify mitigation measures that could be implemented with resources already existing or readily available at the plant, including strategies for fire-fighting, operations to minimize and mitigate fuel damage, and actions to minimize radiological releases.

Parallel but separate plant-specific studies were carried out by the USNRC and the nuclear industry to identify readily available resources to mitigate damage to spent fuel pools and nearby areas from large explosions and fires. The plant conditions evaluated in these site-specific assessments were beyond design basis. The assessments utilized a threat-independent methodology to identify potential plant-specific strategies for preventing or mitigating damage to reactors and spent fuel pools. As described in the industry-sponsored assessment report NEI 06-12[10] (NEI, 2009), the overall strategy involves a diverse capability within plants to provide at least 500 gallons per minute (gpm) of makeup water to the plant's spent fuel pools for 12 hours.[11] The balance of the strategy involves the use of a portable spent fuel pool makeup capability as well as a 200 gpm spray capability from that same water source to enhance the robustness and flexibility of site responses.

The Fukushima Daiichi accident renewed and heightened interest in the potential vulnerability of spent fuel pools to extreme natural events (see Chapter 2 of this report). The USNRC issued Order EA-12-049 (USNRC, 2012a), which directed nuclear plant licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following a beyond-design-basis event. The industry responded with the Diverse and Flexible Coping Strategies (FLEX) initiative (NEI, 2012; see also NRC, 2014, Appendix F). The USNRC subsequently endorsed the industry's FLEX initiative for meeting the Order (see USNRC [2016] for the latest guidance). The FLEX initiative is designed to increase defense-in-depth for beyond-design-basis accident scenarios, including the extended loss of AC power and ultimate heat sink at multiple units on a site.

The objectives of FLEX are to establish resources and associated procedures for an indefinite coping capability to prevent damage to fuel in the

[10] NEI 06-12 is USNRC-endorsed guidance for meeting the requirements of 10 CFR 50.54(hh)(2).

[11] This makeup capability can be used to replace water lost from the pool due to breaches in the pool boundary or from evaporation. (A 500-gpm flow rate approximately corresponds to the drainage rate resulting from an 8-cm-diameter hole in the pool wall 5 m below the pool's free surface.) The water makeup requirements for evaporative losses are well below 500 gpm. For example, only about 40 gpm of water would have been required to remove decay heat in the Unit 4 pool at the Fukushima Daiichi plant 10 days after the entire core was offloaded to the pool.

reactor and spent fuel pool and maintain containment functions.[12] The operator would first use installed equipment, if available, to meet these goals. If such equipment is not available, then operators would provide makeup water (e.g., from the condensate storage tank) with a portable injection source (pump, flexible hoses to standard connections, and associated diesel engine-generator) that can provide at least 500 gpm of spent fuel pool makeup. The portable equipment would be staged on site and could also be brought in from regional staging facilities.

If pool water levels cannot be maintained above the tops of the fuel assemblies, then portable pumps and nozzles would be used to spray water on the uncovered fuel assemblies. FLEX requires a minimum of 200 gpm to be sprayed onto the tops of the fuel assemblies to cool them (NEI, 2012).[13] These FLEX water flow requirements are consistent with the requirements in 10 CFR 50.54(hh)(2). The installed FLEX equipment has also been shown to have substantial margin (60 percent) above the required flow based on equipment testing.

Although good progress has been made by the USNRC and the U.S. nuclear industry in implementing Recommendation 3E-2 in NRC (2006), the committee recommends two additional actions (**Recommendation 4.8**; see Table 4.1 in Chapter 4) be taken:

- *Reexamine the need for the 60-day limit for fuel dispersion after reactor shutdown and reduce the allowable time if feasible.* It is not clear what the technical basis is for this 60-day time period. Not all spent fuel is air-coolable within 60 days of reactor shutdown. The USNRC's Spent Fuel Study (described in Chapter 7 of this report), for example, shows that spent fuel cannot be air cooled for about the first 30 days after its offload from a reactor regardless of fuel configuration in the pool. A shorter-than-60-day time limit may provide an improved safety posture if it is operationally feasible and does not increase the risk of other types of spent fuel pool accidents. A risk-informed examination of this issue could have safety benefits.
- *Reexamine and, if needed, redesign the water makeup and spray systems and strategies to ensure that they can be implemented when physical access to pools is hindered, for example, by structural damage and/or radiation levels or the site becomes inaccessible.* The FLEX strategy for spent fuel pool cooling assumes that

[12] Dry-cask storage facilities are outside the scope of FLEX.

[13] USNRC (2016) notes that the FLEX portable spray capability (utilizing portable spray nozzles from the refueling floor with portable pumps) is not required when a pool is located below grade or when a seismic hazard analysis shows that the pool will maintain its integrity.

workers will have physical access to the pools to install hoses and spray nozzles if permanently installed equipment is damaged. However, physical access might not be possible if the building is damaged or the pool is drained (in the latter case, high radiation levels would likely limit physical access to the pool). The spent fuel pools in Units 1-4 of the Fukushima Daiichi plant were not accessible after the hydrogen explosions because of debris and high radiation levels.

APPENDIX 6A
USNRC and Sandia Studies on Loss-of-Cooling Accidents in Spent Fuel Pools

This Appendix describes a series of USNRC-sponsored and Sandia-executed technical studies that were carried out to improve the understanding of loss-of-cooling accidents in spent fuel pools. These studies were initiated following completion of NRC (2004) and were published between 2006 and 2008. The publications have not been released to the public because they contain security-related information. However, the committee was provided copies of these publications by the USNRC and also received detailed briefings from the USNRC and Sandia on some of this work.

Summaries of these technical studies are provided below to demonstrate the breadth of information that committee considered outside of publicly available documents. These summaries omit technical details that are considered by the USNRC to be security sensitive.

Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 2, Completed November 2006; 121 pages.

Simulations were carried out to examine mitigation strategies for representative spent fuel pools in PWR and BWR reference plants. The following mitigation strategies were examined: makeup water, pool leak repair, fuel dispersion, emergency sprays, building ventilation, and pool configuration.

Analysis of BWR Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Supplemental Air Cases
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 3, Completed January 2008; 63 pages.

This study is a follow-up to earlier studies (e.g., NUREG-1726), which used computational fluid dynamics (FLUENT) to simulate air-flow patterns in drained spent fuel pools. This study used FLOW 3-D to model natural air convection in a fully drained BWR (Mark I) spent fuel pool. A porous media model for the fuel racks and fuel was used together with a computational fluid dynamics model of the pool and reactor building. Air-flow patterns within and outside the fuel were simulated. The effect of open areas in the pool (i.e., areas of the pool with no racking) on peak air temperatures was examined in a series of parametric computations. Fuel oxidation or mechanical response was not modeled, so no conclusions were reached regarding fuel damage or zirconium fires.

Analysis of Emergency Spray Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 2, Completed January 2008; 115 pages.

MELCOR was used to simulate water sprays in mitigating loss-of-coolant accidents in a BWR (Mark I) spent fuel pool. Parametric studies were performed to examine the effects of pool leak location and size, spray flow rate, and fuel arrangement pattern (uniform, 1 × 4, and checkerboard; see Figure 7.2 in Chapter 7). Steam and air oxidation as well as hydrogen generation and combustion were modeled. The time to fuel ignition was determined as a function of fuel age (i.e., time since reactor shutdown). Scenarios with water levels above and below the baseplate of the fuel racks were considered. Experimental measurements of pressure drops in a prototypic BWR fuel assembly were used to provide realistic correlations for flow resistance within the fuel bundle.

Additional MELCOR Analyses of BWR Spent Fuel Pool Assembly Accident Response
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 3, Completed June 2008; 206 pages.

MELCOR was used to simulate full and partial loss-of-coolant accidents in a spent fuel pool in a reference BWR (Mark I) plant. Several improvements were made to MELCOR for these analyses, including increasing the fidelity of the rack model, oxidation kinetics, a hydraulic resistance model, and the physical modeling of the fuel assembly and racks. A parametric study of decay heat, "bypass" flow, oxidation-layer thickness, and fuel configuration, including checkerboard, 1 × 4, and uniform configuration cases (see Figure 7.2 in Chapter 7), was performed. Parametric computations were carried out to determine the effect of fuel age and distribution of fuel in the pool on coolability under complete- and partial-loss-of-coolant conditions. The time until fuel ignition (i.e., runaway zirconium oxidation) was quantified for individual assemblies, and source-term computations were carried out for the entire pool. The ability to cool a debris bed with water and air were investigated. The effect of removing the BWR fuel assembly channel on ignition time was also investigated. A simplified analysis of fuel ballooning was carried out.

Evaluation of a BWR Spent Fuel Pool Accident Response to Loss-of-Coolant Inventory Scenarios Using MELCOR 1.8.5
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 3, Completed June 2008; 172 pages (Part I), 175 pages (Part II).

MELCOR was used to examine full- and partial-loss-of-pool-coolant accidents in a BWR (Mark I) reference plant when there is no mitigation and building ventilation is lost. Parameters examined included decay heat, radial thermal coupling scheme, the leakage hole size (small, medium, and large), and an open (i.e., blowout panel removed) versus a closed reactor building. The effect of hydrogen combustion was investigated for the partial-loss-of-pool-coolant cases. Radioactive material releases from the stored fuel were estimated.

Evaluation of a PWR Spent Fuel Pool Accident Response to Loss-of-Coolant Inventory Scenarios Using MELCOR 1.8.5
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 4, Completed June 2008; 156 pages.

MELCOR was used to simulate complete- and partial-loss-of-coolant accidents in a PWR spent fuel pool. For complete loss of coolant, the peak cladding temperature history depends on radial thermal coupling scheme, fuel age, building ventilation rate, and flow resistance of the fuel assemblies. The effect of leak size, crushing of fuel, rod ballooning, and reduced radial thermal coupling were examined through sensitivity calculations. A few simulations were used to compute the magnitude of fission product releases outside the spent fuel pool building.

MELCOR 1.8.5 Separate Effects Analyses of PWR Spent Fuel Pool Assembly Accident Response
K.C. Wagner and R.O. Gauntt, SANDIA Letter Report, Revision 4, Completed June 2008; 100 pages, +100 pages for part 2.

MELCOR was used to examine the effects of varying the following parameters on peak cladding temperatures within fuel assemblies of a single PWR spent fuel pool: decay heat, gas speed and temperature, oxide-layer thickness, flow resistance, rod ballooning, oxidation kinetics, rack configuration, and water level. For multiple fuel assemblies, the effects of fuel dispersion, including 1 × 4 and checkerboard (see Figure 7.2 in Chapter 7), and the location of empty cells were examined. For full pools, the effects of drain-down time on peak cladding temperature history were also examined. The effect of each of these parameters on the peak cladding temperature history was ranked in order of impact. The minimum fuel age required to prevent ignition was determined for each fuel configuration.

Investigations of Zirconium Fires during Spent Fuel Pool LOCAs: PWR Assemblies
Division of System Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission; June 24, 2013; presentation to committee; 23 pages.

This presentation covered the results of Sandia experiments on PWR bundles. Testing used prototypic assemblies to measure hydraulic resistance to air flow to calibrate the MELCOR model used to model air cooling of fully drained pools. Internal electrical heating was used to simulate decay heat with low power and tests were carried out to determine fuel bundle temperatures and air-flow rates as a function of time prior to ignition. Tests were carried out at higher power levels that resulted in ignition. Measurements were made of flow rate, peak cladding temperatures, oxygen concentration in the exit flow, and time to ignition. Tests were carried out to simulate both the 1 × 4 and uniform loading arrangements that are used for spent fuel in pools. MELCOR simulations had been completed for single-assembly tests and were in progress for testing simulating multiple assemblies. Portions of the Sandia work were carried out in the context of an Organisation for Economic Co-operation and Development/Nuclear Energy Agency (NEA) Sandia Fuel Project and some of the results are now available publicly in NEA (2015).

7

Expedited Transfer of Spent Fuel from Pools to Dry Casks

This chapter provides supporting information for the present committee's Finding 4.11 and Recommendation 4.11 in Chapter 4 (see Table 4.1).

FINDING 4.11: The U.S. Nuclear Regulatory Commission has completed a technical analysis of spent fuel pool accident consequences to inform a regulatory decision on expedited transfer of spent fuel from pool to dry cask storage. The analysis was carried out in accordance with prescribed U.S. Nuclear Regulatory Commission regulatory guidance and provides valuable technical information about the impacts of various accident scenarios on spent fuel storage in pools. However, the analysis did not consider spent fuel storage sabotage risks, dry cask storage risks, or certain health consequences that would likely result from a severe nuclear accident. The analysis also used simplifying bounding assumptions that make it technically difficult to assign confidence intervals to the consequence estimates or make valid risk comparisons. A risk assessment that evaluates the three questions of the risk triplet and that accounts for uncertainties in both probability and consequence estimates is needed to address Finding 4E in NRC (2006) to determine whether "earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants."

RECOMMENDATION 4.11: The U.S. Nuclear Regulatory Commission should perform a spent fuel storage risk assessment to elucidate the risks and potential benefits of expedited transfer of spent fuel from

pools to dry casks. This risk assessment should address accident and sabotage risks for both pool and dry storage. The sabotage risks should be assessed using the methodology developed in response to the present committee's Recommendation 4.1B.

These findings and recommendations arise from the committee's reevaluation of Finding 4E from the National Research Council (NRC) report *Safety and Security of Commercial Spent Fuel Storage* (NRC, 2006):

FINDING 4E of NRC (2006): Depending on the outcome of plant-specific vulnerability analyses described in the committee's classified report, the Nuclear Regulatory Commission might determine that earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants.

This chapter is organized into four sections. Sections 7.1-7.3 describe the recent U.S. Nuclear Regulatory Commission (USNRC) analyses that were undertaken to assess the need for early transfer of spent fuel from pools to dry casks at U.S. nuclear plants. Section 7.4 provides supporting discussion for the committee's finding and recommendation.

7.1 BACKGROUND ON USNRC SPENT FUEL STORAGE ANALYSES

Following the 2011 Fukushima Daiichi accident in Japan, the USNRC staff began a systematic review of the agency's procedures and regulations to determine whether improvements were warranted (USNRC NTTF, 2011). This review is described in the committee's phase 1 report (NRC, 2014, see especially Appendix F). The USNRC staff identified spent fuel transfer from pools to dry cask storage as having "a clear nexus to the Fukushima Daiichi event that may warrant regulatory action . . ." (USNRC, 2011b, p. 5) and subsequently determined that further study of storage arrangements was warranted (USNRC, 2012c).

A few months after the Fukushima accident, USNRC staff initiated the Spent Fuel Pool Study, which examined the consequences of a beyond-design-basis earthquake on a spent fuel pool that is similar in design to some of the pools at the Fukushima Daiichi plant (USNRC, 2014a).[1] The primary objective of this study was to

[1] The study was completed in October 2013 and published in September 2014 following a public comment period.

> ". . . determine if accelerated transfer of spent fuel from the spent fuel pool to dry cask storage provides a substantial safety enhancement for the reference plant. The insights from this analysis will inform a broader regulatory analysis of the SFPs [spent fuel pools] at U.S. nuclear reactors as part of the Japan Lessons-learned Tier 3 plan." (USNRC, 2014a, p. 3)

The objective of this broader regulatory analysis, hereafter referred to as the Expedited Transfer Regulatory Analysis, was to determine whether "additional studies are needed to further assess potential regulatory action on expedited transfer" (USNRC, 2013, p. iii). The Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis are described in Sections 7.2 and 7.3 below.

Spent fuel pools at U.S. nuclear plants were originally outfitted with "low-density" storage racks that could hold the equivalent of one or two reactor cores of spent fuel.[2] (See Appendix 7A for a discussion of spent fuel pool racking.) This capacity was deemed adequate because plant operators planned to store spent fuel only until it was cool enough to be shipped offsite for reprocessing. However, reprocessing of commercial spent fuel was never implemented on a large scale in the United States; consequently, spent fuel has continued to accumulate at operating nuclear plants.

U.S. nuclear plant operators have taken two steps to manage their growing inventories of spent fuel. First, "high-density" spent fuel storage racks have been installed in pools to increase storage capacities. This action alone increased storage capacities in some pools by up to about a factor of 5 (USNRC, 2003). Second, dry cask storage has been established to store spent fuel that can be air cooled.[3] Typically, transfers of the oldest (and therefore coolest) spent fuel from pools to dry casks are made only when needed to free up space in the pool for offloads of spent fuel resulting from reactor refueling operations.

The objective of *accelerated* or *expedited transfer* would be to reduce the density of spent fuel stored in pools:

> "Expedited transfer of spent fuel into dry storage involves loading casks at a faster rate for a period of time to achieve a low density configuration in the spent fuel pool (SFP). The expedited process maintains a low density pool by moving all fuel cooled longer than 5 years out of the pool." (USNRC, 2014a, p. B-1)

[2] Additionally, U.S. nuclear plant operators maintain enough open space in their spent fuel pools to offload entire reactor cores if needed for safety or maintenance actions.

[3] All but three operating nuclear plants in the United States (Three Mile Island, Pennsylvania; Shearon Harris, North Carolina; and Wolf Creek, Kansas) have established or are in the process of establishing dry cask storage facilities. See http://pbadupws.nrc.gov/docs/ML1507/ML15078A414.pdf.

The low-density configuration achieved by expedited transfer would reduce inventories of spent fuel stored in pools. This might improve the coolability of the remaining fuel in the pools if water coolant was lost or if cooling systems malfunctioned.

7.2 SPENT FUEL POOL STUDY

The Spent Fuel Pool Study analyzed the consequences of a beyond-design-basis earthquake on a spent fuel pool at a reference plant[4] containing a General Electric Type 4 boiling water reactor (BWR) with a Mark I containment.[5] The USNRC describes this study as one in a continuing series of examinations of postulated spent fuel pool accidents (see Sidebar 7.1).

The USNRC selected an earthquake having an average occurrence frequency of 1 in 60,000 years and a peak ground acceleration of 0.5-1.0 *g* (average 0.7 *g*) as the initiating event for this analysis.[6] The study examined the effects of the earthquake on the integrity of the spent fuel pool and the effects of loss of pool coolant on its stored spent fuel. The scenarios considered in the analysis are summarized in Figure 7.1.

A modeling analysis was carried out to identify initial damage states to the pool structure from this postulated seismic event. The analysis concluded that structural damage to the pool leading to water leaks (i.e., tears in the steel pool liner and cracks in the reinforced concrete behind the liner) was most likely to occur at the junction of the pool wall and floor. This leak location would result in complete drainage of the pool if no action was taken to plug the leak or add make-up water. Given the assumed earthquake, the leakage probability was estimated to be about 10 percent (see upper part of Figure 7.1), which the USNRC staff judged to be conservative.

The Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code (see Sidebar 6.1 in Chapter 6) was used to model the consequences of three leak scenarios, two spent fuel loading configurations in the pool, five reactor operating cycle phases, and two mitigation actions (see Figure 7.1):

Leak scenarios:

- "No leak" in the spent fuel pool.

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[4] The reference plant was Unit 3 of the Peach Bottom nuclear plant in Pennsylvania. It is similar in design to the Fukushima Daiichi Unit 1-5 reactors and spent fuel pools.

[5] Chapter 2 of NRC (2014) discusses reactor designs. See especially Figure 2.5.

[6] The USNRC selected an earthquake as the initiating event for this analysis because its previous studies (Collins and Hubbard, 2001; Throm, 1989) concluded that "seismic events were the largest contributor to the frequency of fuel uncovery" (USNRC, 2014a, p. 8).

SIDEBAR 7.1
Spent Fuel Pool Storage Analyses

The USNRC and its national laboratories have carried out several analyses of spent fuel pool storage at commercial nuclear plants during the past three or so decades. In 1989, Brookhaven National Laboratory carried out a value impact analysis of spent fuel storage at nuclear plants in operation as of 1986 (Jo et al., 1989). It examined the cost effectiveness of several potential safety enhancements: replacing the high-density storage in pools with low-density storage, installation of water sprays, and redundant cooling or makeup systems. None of these options was found to be cost effective, and the main conclusion of the analysis was that plant licensees needed to take care in handling casks to avoid dropping them into pools.

Throm (1989) evaluated the need for additional protective or mitigative measures for improving the safety of spent fuel storage in high-density storage racks in pools at light-water reactor plants. The following seven protective or mitigative measures were specifically evaluated: take no action, require use of low-density storage racks, improve pool cooling/water makeup systems, install water spray systems, modify spent fuel storage rack designs, cover the fuel debris in a drained pool with solid materials, and improve ventilation gas treatment systems. The USNRC carried out a value-impact analysis on each alternative. All of the alternatives considered had substantial negative net benefits, and the USNRC's safety goals (Sidebar 7.2) were met without implementing any of the alternatives. The recommendation from the regulatory analysis was to take no action.

Spent fuel pool safety in decommissioning reactors was addressed by Collins and Hubbard (2001). This study considered a wide range of initiating events and accident sequences with limited consequence analyses. No cost-benefit analyses were performed. The report concluded that

> "Deterministic evaluations in the staff's preliminary draft risk assessment indicated that zirconium cladding fires could not be ruled out for loss of SFP [spent fuel pool] cooling for fuel that has been shut down and removed from an operating reactor within approximately 5 years. The consequence analysis indicated that zirconium cladding fires could give offsite doses that the [US]NRC would consider unacceptable." (pp. A2-1–A2.2)

There were substantial uncertainties regarding the source term from oxidation of fuel in air. Although the effects of fuel loading density and dispersal of fuel assemblies in the pool were not considered quantitatively, there was a clear recognition of the importance of these factors:

> "Prediction of the propagation of the temperature escalation to the cooler regions of the pool needs to be carefully examined to see if significant benefit can be gained, at a minimum it will lengthen the period of fission product release reducing the concentration of activity in the plume of fission products for offsite consequence analysis." (Collins and Hubbard, 2001, pp. A1B-7–A1B-8)

continued

SIDEBAR 7.1 Continued

The overall conclusion of Collins and Hubbard (2001) was that

> "The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious" (p. 5-3).

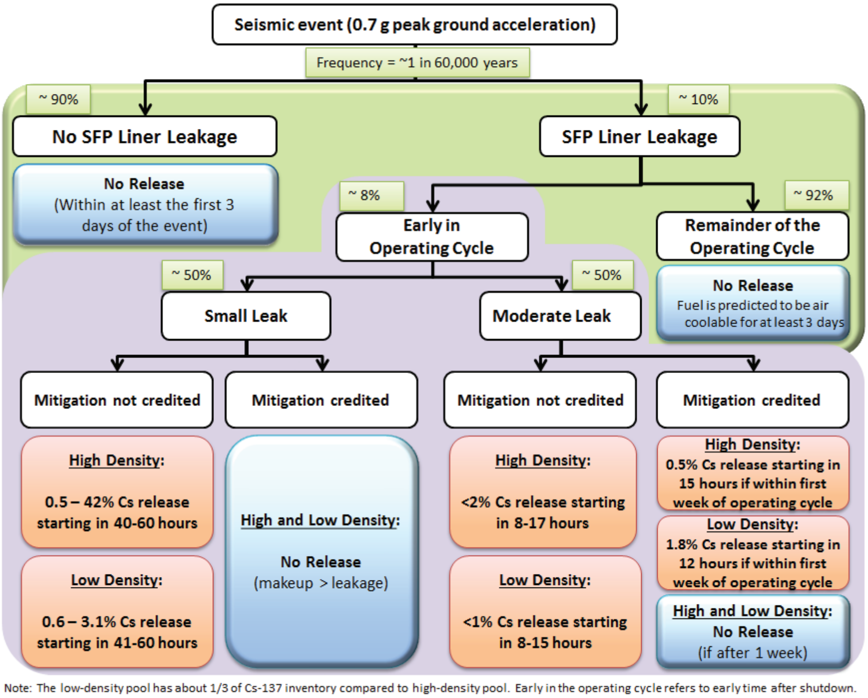

FIGURE 7.1 Scenarios analyzed in the Spent Fuel Pool Study. Note: SFP = spent fuel pool; Cs = cesium-137. SOURCE: USNRC (2014a, Figure ES-1).

- A "small leak" in the pool that averages about 200 gallons per minute for water heights at least 16 feet above the pool floor (i.e., at the top of the spent fuel rack).
- A "moderate leak" in the pool that averages about 1,500 gallons per minute for water heights at least 16 feet above the pool floor.

Reactor operating cycle phases:[7]

- OCP1: 2-8 days; reactor is being defueled.
- OCP2: 8-25 days; reactor is being refueled.
- OCP3: 25-60 days; reactor in operation.
- OCP4: 60-240 days; reactor in operation.
- OCP5: 240-700 days; reactor in operation.

Fuel configurations in the pool:[8]

- A "high-density" storage configuration in which hot (i.e., recently discharged from the reactor) spent fuel assemblies are surrounded by four cooler (i.e., less recently discharged from the reactor) fuel assemblies in a 1 × 4 configuration throughout the pool (Figure 7.2).
- A "low-density" storage configuration in which all spent fuel older than 5 years has been removed from the pool.[9]

Mitigation scenarios:

- A "mitigation" case in which plant operators are successful in deploying equipment to provide makeup water and spray cooling required by 10 CFR 50.54(hh)(2)[10] (see Chapter 2).
- A "no-mitigation" case in which plant operators are not successful in taking these actions.

Some key results of the consequence modeling are shown in Table 7.1 and summarized in the bottom panels of Figure 7.1. Some of the loss-of-coolant scenarios examined in the study resulted in damage to, and the release of, radioactive material from the stored spent fuel. Releases began

[7] Phases in a 2-year cycle of removing (defueling) one-third of the reactor core and placing that spent fuel into the pool. The phases are referenced to the start of the defueling cycle.

[8] Sensitivity analyses were also carried out for uniform and 1 × 8 storage configurations (Figure 7.2) to determine the effect of fuel configuration on the consequence estimates.

[9] The high-density racking remained in the pool for this scenario; other racking scenarios that would allow for lateral water flow across the fuel racks (i.e., "open racking" scenarios) were not considered.

[10] Available mitigation equipment includes portable pumps, hoses, and spray nozzles.

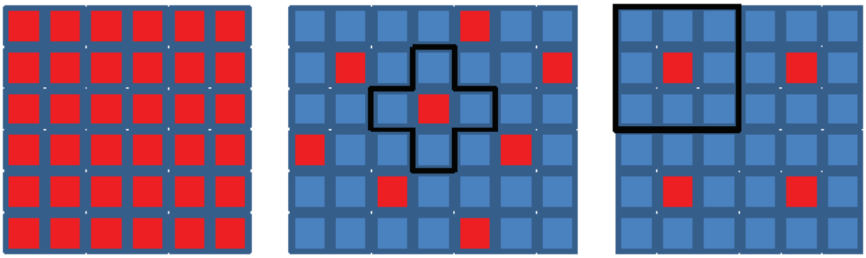

FIGURE 7.2 Spent fuel dispersion patterns considered in the Spent Fuel Pool Study. Notes: Each square represents a spent fuel assembly in a spent fuel pool rack. Red fuel assemblies have higher heat decay. Blue fuel assembles have lower heat decay because the fuel was discharged in an earlier refueling cycle. The black border in each image denotes the repeat patterns. Left image, uniform dispersion; middle image, 1 × 4 dispersion; right image, 1 × 8 dispersion. SOURCE: USNRC (2014a, Figure 34).

TABLE 7.1 Key Results from the Consequence Analysis in the Spent Fuel Pool Study

Spent Fuel Pool Fuel Loading	High Density (1 × 4) (Regulatory Baseline)		Low Density (Proposed Alternative)	
Seismic hazard frequency[a] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05		1.7E-05	
50.54(hh)(2) mitigation credited?	Yes	No	Yes	No
Conditional[b] probability of release (%)	0.036	0.69	0.036	0.69
Hydrogen combustion event?	Not Predicted	Possible	Not Predicted	Not Predicted
Conditional[c] consequences (release frequency-averaged[d])				
Cumulative Cs-137 release at 72 hours (MCi)	0.26	8.8[h]	0.19[g]	0.11
Measures related to health and safety of individuals				
Individual early fatality risk	0	0	0	0
Individual latent cancer fatality risk[e] Within 10 miles of plant	3.4E-04	4.4E-04	3.4E-04	2.0E-04
Measures related to cost-benefit analysis				
Collective dose (person-Sv) Within 1,000 miles of plant[i]	47,000	350,000	47,000	27,000
Land interdiction[f] (mi^2) Within 1,000 miles of plant[i]	230	9,400	230	170
Long-term displaced individuals[f] Within 1,000 miles of plant[i]	120,000	4,100,000	120,000	81,000

TABLE 7.1 Continued

Spent Fuel Pool Fuel Loading	High Density (1 × 4) (Regulatory Baseline)		Low Density (Proposed Alternative)	
Consequences per year (release frequency-weighted[d])				
Release frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
Measures related to health and safety of individuals				
Individual early fatality risk (/yr)	0	0	0	0
Individual latent cancer fatality risk[e] Within 10 miles of plant (/yr)	2.1E-12	5.2E-11	2.1E-12	2.4E-11
Measures related to cost-benefit analysis				
Collective dose (person-Sv/yr) Within 1,000 miles of plant[i]	2.9E-04	4.1E-02	2.9E-04	3.2E-03
Land interdiction[f] (mi^2/yr) Within 1,000 miles of plant[i]	1.4E-06	1.1E-03	1.4E-06	2.0E-05
Long-term displaced individuals[f] (persons/yr) Within 1,000 miles of plant[i]	7.1E-04	4.9E-01	7.1E-04	9.5E-03

NOTE: The individual early fatality risk estimates and individual latent cancer fatality risk estimates shown in the table were not derived from a risk assessment. They were computed using the postulated earthquake and scenario frequencies shown in the table. PGA = peak ground acceleration.
[a] Seismic hazard model from Petersen et al. (2008).
[b] Given that the specified seismic event occurs.
[c] Given atmospheric release occurs.
[d] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions (as applicable); additionally, "release frequency-weighted" results are multiplied by the release frequency.
[e] Linear no-threshold and population weighted (i.e., total amount of latent cancer fatalities predicted in a specified area, divided by the population that resides within that area).
[f] First year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51.
[g] Mitigation can moderately increase release size; the effect is small compared to the reduction in release frequency.
[h] Largest releases here are associated with small leaks (although sensitivity results show large releases are possible from moderate leaks). Assuming no complications from other spent fuel pools/reactors or shortage of available equipment/staff, there is a good chance to mitigate the small leak event.
[i] Kevin Witt, USNRC, written communication, December 22, 2015.

SOURCE: USNRC (2014a, Table 33).

anywhere from several hours to more than 2 days after the postulated earthquake. The largest releases were estimated to result from high-density fuel storage configurations with no mitigation (Figure 7.1). The releases were estimated to be less than 2 percent of the cesium-137 inventory of the stored fuel for medium-leak scenarios, whereas releases were estimated to be one to two orders of magnitude larger for small-leak scenarios with a hydrogen combustion event. Hydrogen combustion was found to be "possible" for high-density pools but "not predicted" for low-density pools.

Operating-cycle phase (OCP) played a critical role in determining the potential for fuel damage and radioactive materials release. The potential for damage is highest immediately after spent fuel is offloaded into the pool (OCP1) because its decay heat is large. The potential for damage decreases through successive operating-cycle phases (OCP2-OCP5). In fact, only in the first three phases (OCP1-OCP3) is the decay heat sufficiently large to lead to fuel damage in the first 72 hours after the earthquake for complete drainage of the pool. These three "early in operating cycle" phases (Figure 7.1) constitute only about 8 percent of the operating cycle of the reactor.

A limited-scope human reliability analysis (HRA) was conducted to estimate the likelihood of successful operator actions to prevent spent fuel damage following the earthquake (USNRC, 2014a, Chapter 8). The analysis estimated that the probability of failure to successfully complete required mitigating actions was highest in OCP1-OCP3, particularly for moderate-leak scenarios. (The probability of failure of successful mitigation was estimated to be 1 in the case of a moderate leak in both OCP1 and OCP3.) The probability of failure of mitigating action ranged from 0.15 to 0.75 for the moderate-leak scenarios in OCP2, depending on whether a station blackout with and without DC power is assumed (USNRC, 2014a, Table 49). The HRA suggests that "no mitigation" is a prudent assumption for moderate-leak scenarios in OCP1-OCP3.[11]

The USNRC summarized the results of the consequence analysis as follows:

[11] Chapter 8 of USNRC (2014a) notes that the human-error probabilities were estimated under the assumption that mitigation equipment was available, there was no simultaneous core damage or primary containment failure that caused inaccessibility of the refueling floor, and there was sufficient staff to deploy the spent fuel pool mitigation strategy. If the earthquake caused damage in multiple reactors and spent fuel pools, such as occurred at the Fukushima Daiichi plant (see Chapter 2 of this report), then these assumptions might not hold. The authors indicate that examination of these additional considerations would require performance of a more comprehensive probabilistic risk assessment and associated HRA. The more comprehensive risk analysis recommended by the present committee (Recommendation 4.11) would be able to address the impact of these types of considerations on the likelihood of successful mitigating actions.

> ". . . in a high-density loading configuration, dispersing hotter fuel throughout the pool or successful mitigation generally prevented or reduced the size of potential releases [of radioactive material from stored spent fuel]. Low-density loading reduced the size of potential releases but did not affect the likelihood of a release. When a release is predicted to occur, early and latent fatality risks for individual members of the public do not vary significantly between the scenarios studied because protective actions, including relocation of the public and land interdiction,[12] were modeled to be effective in limiting exposure. The beneficial effects in the reduction of offsite consequences between a high-density loading scenario and a low-density loading scenario are primarily associated with the reduction in the potential extent of land contamination and associated protective actions." (USNRC, 2014a, p. xxix)

The Spent Fuel Pool Study (USNRC, 2014a, Appendix D) also included a regulatory analysis for the reference plant. This entailed a comparison of the consequences to the public from the postulated releases from the reference plant against the quantitative health objectives (QHOs) in the USNRC's Safety Goal Policy Statement (Sidebar 7.2) and also the development of a cost-benefit analysis to determine whether expedited transfer is cost beneficial.[13]

The USNRC estimated that releases from the reference plant would not result in any early fatalities from acute radiation exposures within 1 mile (1.6 kilometers) of the plant boundary (QHO 1 in Sidebar 7.2). The USNRC also estimated the individual latent cancer fatality risk from the accident within a 10-mile (16-kilometer) radius of the plant boundary to be in the range 10^{-11}–10^{-12} (Table 7.1). This is a miniscule risk—about six orders of magnitude lower than the 2×10^{-6} per year objective (QHO 2 in Sidebar 7.2).

The USNRC summarized the results of these analyses for the reference plant as follows:

> ". . . expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. . . . The [US]NRC continues to believe, based on this study and previous studies that high density storage of spent fuel in pools protects public health and safety." (USNRC, 2014a, p. xxix)

[12] "Interdiction is the temporary relocation of the affected population while decontamination, natural weathering, and radioactive decay reduce the contamination levels" (USNRC, 2013, p. 103).

[13] That is, to determine whether the benefits of the proposed regulatory action equal or exceed its costs.

SIDEBAR 7.2
Safety Goal Policy Statement

The USNRC's Safety Goal Policy Statement (USNRC, 1986) contains two qualitative safety goals and two quantitative health objectives:

Qualitative Safety Goals

1. Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
2. Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

Quantitative Health Objectives

1. The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities[a] that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks results from other accidents to which members of the U.S. population are generally exposed. The QHO is 5×10^{-7} per year for an average individual within 1 mile (1.6 kilometers) of the plant site boundary (USNRC, 1983).
2. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes. The QHO is 2×10^{-6} per year for an average individual within 10 miles (16 kilometers) of the plant site boundary (USNRC, 1983).

Safety goals are used to screen potential regulatory actions that are considered to be generic safety enhancements to nuclear plants under the backfit regulation (10 CFR 50.109; see Sidebar 5.5 in NRC [2014]). The Expedited Transfer Regulatory Analysis described in this chapter is an example of a regulatory action that is subject to safety goal screening.

Some analysts have proposed the addition of one or more societal-risk QHOs to account for the probabilistically weighted risks of large-scale evacuations and large population doses (Bier et al., 2014; Denning and McGhee, 2013).

[a] A *prompt fatality* is a death that occurs within a year following the accidental release of radioactive material as a result of radiation exposures received during the first week of that release.

7.3 EXPEDITED TRANSFER REGULATORY ANALYSIS

A second analysis, the Expedited Transfer Regulatory Analysis, extended the Spent Fuel Pool Study to the fleet of commercial spent fuel pools and independent spent fuel storage installations in the United States. The analysis considered two storage alternatives:

- A *regulatory baseline alternative* that maintains existing storage requirements, that is, "storage of spent fuel in high-density racks in the SFP [spent fuel pool], a relatively full SFP, and compliance with all current regulatory requirements" (USNRC, 2013, p. 5). As noted in Chapter 6 of this report, current regulations require that fuel be dispersed in a 1 × 4 pattern of high- and low-decay-heat assemblies (see Figure 7.2) following each fuel offload from the reactor not later than 60 days after reactor shutdown, unless such configuration can be shown to be inapplicable or unachievable.
- A *proposed alternative* that "would require older spent fuel assemblies to be expeditiously moved from SFP storage to dry cask storage beginning in year 2014, to achieve and maintain a low-density loading of spent fuel in the existing high-density racks as a preventive measure" (USNRC, 2013, p. 6). The USNRC identified three benefits of this alternative: less long-lived radionuclide inventory in the spent fuel pool, lower heat load in the pool, and a small increase in the initial water inventory in the pool (because water would displace the fuel assemblies that were moved from the pool to dry cask storage).

The analysis utilized seven groupings of nuclear plants and their spent fuel pools based on "conservative estimates and assumptions to bound the variations in SFP parameters across the fleet . . ." (USNRC, 2013, p. iv and Table 1):

1. BWR Mark I and II reactors with nonshared pools (31 reactors, 31 pools),
2. BWR Mark III reactors and pressurized water reactors (PWRs) with nonshared pools (49 reactors, 49 pools),
3. AP1000 reactors,[14]
4. Reactors with shared spent fuel pools (20 reactors, 10 pools),
5. Spent fuel pools located below grade (a subset of the PWR reactors in Group 2),

[14] No AP1000 reactors are currently operating in the United States, but four reactors are currently under construction.

6. Decommissioned reactors with spent fuel pools (7 reactors, 6 pools), and
7. Decommissioned reactors with only dry cask storage (21 reactors, no pools).

The regulatory analysis focused on the first four groups of spent fuel pools. The pools in group 5 were excluded from the analysis because they were below grade and therefore deemed to be "less susceptible to the formation of small or medium leaks due to the absence of open space around the pool liner and concrete structure" (USNRC, 2013, p. 11).

The analysis considered eight types of initiating events that were judged to have the potential to lead to the loss of cooling in spent fuel pools: seismic events, drops of casks and other heavy loads on pool walls, loss of offsite power, internal fire, loss of pool cooling or water inventory, inadvertent aircraft impacts, wind-driven missiles,[15] and failures of pneumatic seals on the gates in the spent fuel pools (USNRC, 2013, Table 43). These initiating events could lead to partial or full drainage of the spent fuel pools. If full drainage occurs, then air cooling of the fuel to prevent its runaway oxidation (i.e., a zirconium cladding fire) was assumed to be feasible 60 days following its discharge from the shutdown reactor. If partial drainage occurs in pools with racks that block natural air circulation, then air cooling was assumed not to be feasible (USNRC, 2014a, p. 18).

The expedited transfer regulatory analysis was carried out in two parts:

1. The potential safety benefits of expedited transfer were screened using the QHOs in the USNRC's Safety Goal Policy Statement (Sidebar 7.2).
2. A cost-benefit analysis was carried out to determine whether expedited transfer would be cost beneficial.

7.3.1 Safety Goal Screening

The pool-weighted averages of release frequencies of fission products to the environment from the seven types of initiating events described previously range between 7.39×10^{-7} and 2.88×10^{-5} per pool per year without successful mitigation (USNRC, 2013, Table 43). Even though some releases were large, the USNRC concluded that they would not result in any early fatalities from acute radiation exposures within 1 mile (1.6 kilometers) of the plant boundary (QHO 1 in Sidebar 7.2). The USNRC estimated

[15] That is, the wind-driven impacts of heavy objects (e.g., storm debris) on external walls of spent fuel pools.

the individual latent cancer fatality risk[16] within a 10-mile (16-kilometer) radius of the plant boundary to be 1.52×10^{-8} per year. This is less than 1 percent of the 2×10^{-6} per year objective (QHO 2 in Sidebar 7.2).

The USNRC staff concluded that

> ". . . the continued operation of nuclear power plants with high-density loadings in their SFPs [i.e., the regulatory baseline alternative] does not challenge the [US]NRC's safety goals or related QHOs. Therefore, in the staff's judgment, a regulatory action to require reducing the inventory of spent fuel in the pools would provide no more than a minor safety improvement." (USNRC, 2013, p. 10)

A regulatory analysis would normally be terminated once USNRC staff determined that the alternative action (in this case expedited transfer) did not provide a sufficient safety enhancement relative to the Commission's safety goals and objectives. However, the USNRC staff performed a cost-benefit analysis even though the computed risks were well below the QHOs. The USNRC staff stated that it provided the cost-benefit analysis "to provide the Commission additional information" (USNRC, 2013, p. 2).

7.3.2 Cost-Benefit Analysis

The cost-benefit analysis considered several attributes that could be affected by the two storage alternatives (i.e., the regulatory baseline alternative and proposed alternative) consistent with the USNRC's regulatory guidance (USNRC, 1997b, 2004) and U.S. Office of Management and Budget (OMB) guidance (1992) (Sidebar 7.3). The attributes are described in Chapter 5 of USNRC (1997b) and are summarized below:

- Public Health[17] (accident and routine). In the base case, changes in estimated exposures of the public to radiation caused by changes in accident frequencies or consequences associated with the alternative action measured over a 50-mile (80-kilometer) radius from the plant site boundary are considered. Exposures can result from continued occupation or reoccupation of radioactively contaminated land following a release from a spent fuel pool as well as worker exposures resulting from cleanup and decontamination of offsite land.

[16] As noted previously, these risks were not derived from a risk assessment but are based on postulated earthquake and scenario frequencies.

[17] The USNRC's public health attributes focus exclusively on exposures of workers and the public to radiation during routine (i.e., normal operating) and accident conditions. They do not include other public health effects such as psychological effects.

SIDEBAR 7.3
Cost-Benefit Analysis

The OMB provides guidance for conducting and drawing conclusions from cost-benefit analyses used in federal programs, including those carried out by the USNRC:

> "The OMB maintains that the regulatory analysis should select the regulatory alternative that achieves the greatest present value in terms of the discounted monetized value of expected net benefits (i.e., benefits minus costs)." (USNRC, 2004, p. 33)

The quantitative value of benefits is computed using the methodology discussed in USNRC (2004). This involves the quantification of the change in "attributes" (see Section 7.3.2) usually through computing mean or expected values. USNRC (2004) provides guidelines for the valuation of 17 specific attributes.

The net benefit is computed as the difference between the sum of the value of all benefits and the total cost to the industry and the USNRC for carrying out the actions required to obtain the benefits. USNRC (2004) specifies how the value analysis should be carried out. A key constraint for assessing the consequences of severe spent fuel pool accidents is the guidance for assessing offsite impacts:

> "In the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site." (USNRC, 2004, p. 29)

Although the USNRC (2004) requires quantification of costs and benefits to the extent possible, it also allows for the incorporation of qualitative factors:

> "Values and impacts that are determined to be unquantifiable should be identified and discussed qualitatively. An attribute should not be omitted from a regulatory analysis document simply because it is determined to be unquantifiable." (p. 24).

The USNRC as yet has no consistent framework for addressing qualitative factors in cost-benefit studies, although the staff is in the process of developing such a framework (USNRC, 2014b). There is a large literature on systematically including qualitative factors in decision tools (see DCLG, 2009; Insua and French,1991; USNRC, 2014b).

- Occupational Health (accident and routine). Two occupational health attributes are considered: (1) changes in exposures of workers to radiation caused by changes in accident frequencies or consequences associated with the alternative action and (2) routine radiological exposures to workers resulting from dry storage cask loading and handling associated with the alternative action.

- Property. Two property attributes are considered: (1) monetary effects on offsite property resulting from radiological releases associated with the alternative action, including direct (e.g., land, food, and water) and indirect (e.g., tourism) effects and (2) monetary effects on onsite property, including replacement power costs[18] and decontamination and refurbishment costs associated with the alternative action.
- Industry. Two industry attributes are considered: (1) net economic effects on nuclear plant licensees resulting from the implementation of any mandated changes associated with the alternative action and (2) net economic effects resulting from recurring operational activities associated with the alternative action.
- USNRC. Two USNRC attributes are considered: (1) net economic effect on the USNRC resulting from implementation of the alternative action and (2) net economic effect from recurring activities (e.g., inspections and enforcement activities) associated with the alternative action.

The USNRC staff developed expected values for each cost and benefit:

> "The expected value is the product of the probability of the cost or benefit occurring and the consequences that would occur assuming the event happens. For each alternative, the staff first determines the probabilities and consequences for each cost and benefit, including the year the consequence is incurred. The [US]NRC staff then discounts the consequences in future years to the current year of the regulatory action for purposes of evaluating benefits and costs (i.e., providing a net present value). Finally, the [US]NRC staff sums the costs and the benefits for each alternative and compares them." (USNRC, 2013, p. 14)

Sensitivity analyses were performed to assess the effects of four factors on the cost-benefit analysis (USNRC, 2013, Table 3):

- Discount rate for determining net present value. 7 percent rate for the base case and 2 percent and 3 percent rates for the sensitivity analysis;
- Averted dose conversion factor. $2,000 per person-rem for the base case and $4,000 per person-rem for the sensitivity analysis;

[18] The cost of replacement power is the difference between the cost of electricity from the shutdown nuclear reactor and the next least-costly available generating source. The Spent Fuel Pool Study assumed that only the nuclear plant where the accident occurred would be taken out of operation. This was not the case in Japan, where all Japanese plants were shut down following the Fukushima Daiichi accident.

- Replacement energy costs. \$2.3 million for the base case and \$729,000 to \$57.3 million for the sensitivity analysis; and
- Consequences extending beyond 50 miles. 50 miles for the base case and beyond 50 miles for the sensitivity analysis.

USNRC staff concluded that

> ". . . the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP [spent fuel pool] storage alternative are not warranted in light of the benefits from such expedited transfer. The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in this analysis result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs. However, even in these cases, there is only a limited safety benefit when using the QHOs and the expected implementation costs would not be warranted. In addition, in the staff's judgment, the various assumptions made in the analysis of the "base case" result in an overall cost-benefit assessment that is appropriately conservative for a generic regulatory decision and justify using the "base case" as the primary basis for the staff's recommendation." (USNRC, 2013, pp. v-vi)

The USNRC staff recommended "that additional studies and further regulatory analyses of this issue [expedited transfer] not be pursued, and that this Tier 3 Japan lessons-learned activity be closed" (USNRC, 2013, transmittal memo, p. 2). A majority of the Commissioners accepted the staff's recommendation.[19] In other words, the Commission decided not to require its licensees to expedite the transfer of spent fuel from pools to dry casks because the Expedited Transfer Regulatory Analysis showed that the costs of such transfer exceeded the benefits.

7.4 DISCUSSION

The USNRC staff put a great deal of thought and effort into the development of the Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis and the explication of their results. The staff also spent a good deal of time presenting the results of these analyses to the committee and responding to follow-up questions. The presentations helped to sharpen the

[19] Commission Voting Record on COMSECY-13-0030. Available at http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2013/2013-0030comvtr.pdf; and Memo from the Secretary of the NRC Commission to the Executive Director of Operations, Staff Requirements–COMSECY-13-0030–Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel. Available at http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2013/2013-0030comsrm.pdf.

committee's understanding of these USNRC analyses and its assessment of their usefulness for reevaluating Finding 4E in NRC (2006).

The Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis are valuable technical contributions to understanding the consequences of spent fuel pool accidents. However, the USNRC's analyses are of limited use for assessing spent fuel storage risks[20] because

1. Spent fuel storage sabotage risks are not considered.
2. Dry cask storage risks are not considered.
3. The attributes considered in the cost-benefit analysis (Section 7.3.2) are limited by OMB and USNRC guidance and do not include some expected consequences of severe nuclear accidents.
4. The analysis employs simplifying bounding assumptions that make it technically difficult to assign confidence intervals to the consequence estimates or make valid risk comparisons.

The present committee's recommended risk analysis (Recommendation 4.11 in Table 4.1) would provide policy makers with a more complete technical basis for deciding whether earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of accidents and terrorist attacks on stored spent fuel. This recommended risk analysis should

- Consider accident and sabotage risks for both pool and dry cask storage.
- Consider societal, economic, and health consequences of concern to the public, plant operators, and the USNRC.
- More fully account for uncertainties in scenario probabilities and consequences.

These points are discussed further in the following sections.

7.4.1 Sabotage Risks

The Expedited Transfer Regulatory Analysis considered a large number of initiators for spent fuel pool *accidents* (see Section 7.3); the analysis did not include initiators for spent fuel pool *sabotage*. The USNRC staff asserted that it was unnecessary to include sabotage initiators because

[20] As noted in Section 7.4.5 of this chapter, USNRC staff characterized the Spent Fuel Pool Study as "a limited-scope consequence assessment that utilizes probabilistic insights" (USNRC, 2014a, p. 6).

> "For nuclear power plants, security requirements are established to provide high assurance of adequate protection from radiological sabotage of the nuclear power plant reactor and SFP [spent fuel pool]. The [US]NRC continually monitors threat conditions and, as was done after the September 11, 2001 attacks, makes adjustments, as appropriate in the governing security requirements and in actions to oversee their effective implementation. Based on the staff's view that security issues are effectively addressed in the existing regulatory program, they are not part of this analysis." (USNRC, 2013, p. v)

The USNRC staff did not provide the committee with a technical analysis to support its assertion that security requirements are being effectively addressed in its regulatory program.[21] Moreover, the staff's approach for handling sabotage risks is logically inconsistent with how it handled accident risks in the Spent Fuel Pool Study and Expedited Transfer Analysis: In those analyses, staff assumed a nonzero probability that regulatory requirements for mitigation capabilities in 10 CFR 50.54(hh)(2) were not effective (the "no-mitigation" case), even though these requirements are a condition of every nuclear plant operating license in the United States.

The Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis considered beyond-design-basis severe accidents where required mitigation actions failed to be completed successfully. A complete analysis would also include similar considerations for sabotage threats, including the consequences should a design-basis-threat (DBT) event fail to be mitigated, as well as the consequences should beyond-DBT events occur and fail to be mitigated. A complete analysis would consider a broad range of potential threats including insider and cyber threats.

Sabotage initiators can differ from accident initiators in important ways: For example, most accident initiators occur randomly in time[22] compared to the operating cycle of a nuclear plant. Sabotage initiating events can be timed with certain phases of a plant's operating cycle, changing the conditional probabilities of certain attack scenarios as well as their potential consequences (Sidebar 7.4). There may be additional differences between accident and sabotage events with respect to timing, severity of physical damage, and magnitudes of particular consequences, for example radioactive material releases.

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[21] USNRC staff provided briefings to the committee on security at U.S. nuclear plants and described several security changes that have been implemented since the September 11, 2001, terrorist attacks on the United States; some of these changes involved spent fuel pools. Most of these changes have not been publicized for security reasons.

[22] For example, earthquake and storm initiators occur randomly in time, although some storm initiators are seasonal (e.g., hurricanes). Cask-drop initiators, on the other hand, would typically occur prior to reactor refueling outages.

SIDEBAR 7.4
Safety versus Sabotage Conditional Risks

The frequency of radioactive material releases from a spent fuel pool, denoted *F(release)* events per reactor year, is determined by multiplying the initiating event frequency by the conditional probabilities (*P*) for each of the sequences of events that led to the release (see Section 5.6 in USNRC [2014a]):

$$F(release) = P(release \mid loss\ coolant) \times P(loss\ of\ coolant \mid operating\ cycle\ vulnerability) \times P(operating\ cycle\ vulnerability \mid loss\ of\ offsite\ power) \times P(loss\ of\ offsite\ power \mid earthquake) \times F(earthquake)$$

The conditional probabilities are estimated on the basis of past work, expert judgment, and the results of deterministic computations using MELCOR to model specific loss-of-pool-coolant and pool loading scenarios (Section 5.6 and Table 74 of USNRC [2014a]). Representative results for the frequencies and probabilities for the "no-mitigation" scenario in the Spent Fuel Pool Study are shown in the second column in Table S7.1.

The initiating frequency and conditional probabilities in the equation are combined through multiplication, implying a lack of correlation in time. This is probably a reasonable assumption for accident initiators involving natural phenomena such as earthquakes, which tend to be random in time. However, the probabilities associated with sabotage initiators are more likely to be correlated.

TABLE S7.1 Initiating Frequencies and Conditional Probabilities for Seismic (middle column) and Sabotage (right-hand column) Initiators

	Seismic Initiator	Sabotage Initiator (fictitious example)	
F(initiator) (per reactor year)	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}
P(loss of offsite power I initiator)	0.84	1.0	.5
P(operating cycle vulnerability I loss of offsite power & initiator)	0.08	1.0	.5
P(loss of coolant I operating cycle & initiator)	0.1	1.0	.5
P(release I loss coolant)	1.0	1.0	1.0
F(release) (per reactor year)	1.1×10^{-7}	1.7×10^{-5}	2.2×10^{-6}

SOURCE: Middle column, USNRC (2014a, Fig ES-2); right-hand column, committee generated.

continued

SIDEBAR 7.4 Continued

To illustrate this point, consider the following version of the equation that is modified to account for sabotage initiators:

$$F(release) = P(release \mid loss\ coolant) \times$$
$$P(loss\ of\ coolant \mid operating\ cycle\ vulnerability\ \&\ sabotage) \times$$
$$P(operating\ cycle\ vulnerability \mid loss\ of\ offsite\ power\ \&\ sabotage) \times$$
$$P(loss\ of\ offsite\ power \mid sabotage) \times F(sabotage)$$

The following three conditional probabilities could have correlated and high numerical values if knowledgeable and determined saboteurs attack the plant in certain ways during certain parts of its operating cycle:

- *P(loss of offsite power | sabotage)*,
- *P(operating cycle vulnerability | loss of offsite power & sabotage)*, and
- *P(liner damage leading to loss of coolant | operating cycle vulnerability & sabotage)*.

If one assumes, for example, that these conditional probabilities are 1.0, then release frequencies will be about two orders of magnitude higher (third column in the table) than those for a seismic initiator. This increased frequency is a consequence of the correlated behavior of the saboteurs with the reactor operating cycle and a high probability of success using a strategy that exploits plant vulnerabilities. On the other hand, decreasing these three conditional probabilities by a factor of 2 (corresponding to either less successful attackers or more successful defenders) will decrease the likelihood of a release by a factor of 10 (right-hand column in table).

Although the conditional probabilities used in the foregoing scenarios are entirely fictitious (and the scenarios themselves are in no way representative of the broad range of scenarios that could be considered), their use illustrates two important points: (1) A large range of *F(release)* outcomes are possible depending on the conditional probabilities used in the analysis, and, therefore, (2) it is essential to characterize the uncertainties in *F(release)* as part of the analysis. A sabotage risk assessment could be used to estimate these outcomes and uncertainties.

The committee judges that it is not technically justifiable to exclude sabotage risks without the type of technical analysis that is routinely performed for assessing reactor accident risks. Such an analysis would consider both design-basis and beyond-design-basis threats. The likelihoods of these threats could be assessed through elicitation of experts who are knowledgeable about the intents and capabilities of potential saboteurs and who have the appropriate personnel security clearances to access sensitive national security information on terrorist threats. See Chapter 5 for a more detailed discussion of sabotage risks and assessments.

7.4.2 Dry Cask Storage Risks

The Spent Fuel Pool Study (USNRC, 2014a) examined previous studies of dry cask storage safety risks and developed an updated analysis, the results of which are shown in that study's Table 68. The USNRC staff concluded that

> "Comparison of this [Spent Fuel Pool] study to dry cask storage studies (NUREG-1864 [USNRC, 2007b] and supplemental analyses from [this report]), indicates that in some circumstances, the conditional individual LCF [latent cancer fatality] risk within 0 to 10 miles would be similar due primarily to the conservative upper bound estimate of the dry cask release as well as the expected effectiveness of protective actions in response to an SFP release. However, conditional results for metrics such as population dose or condemned[23] or interdicted lands are several orders of magnitude lower for dry cask scenarios than the low end of consequences of pool accidents, due to the substantially smaller amount of released material." (USNRC, 2014a, p. 254-255)

The Expedited Transfer Regulatory Analysis did not examine the safety or sabotage risks of dry cask storage. The study did not consider sabotage risks and "conservatively ignored" the risks of handling and loading dry casks to calculate the maximum potential benefits and implementation costs (USNRC, 2013, p. 33). The committee judges that a more in-depth examination of the risks associated with dry-cask storage are needed to fully inform an analysis of spent fuel storage risks. The committee encourages the USNRC to develop a risk assessment that explicitly accounts for the risks associated with both pool and dry cask storage.

7.4.3 Expected Consequences

The USNRC uses safety goal screening (Sidebar 7.2) in all of its regulatory analyses that impact nuclear plant licensees to determine whether a substantial safety enhancement exists. The safety goals used in this screening were developed following the 1979 accident at the Three Mile Island (TMI) plant in Pennsylvania. The goals and objectives were influenced substantially by the characteristics of that accident, which involved modest leakage of radioactive material from the reactor's containment. The possibility of spent fuel pool accidents was recognized but largely unanalyzed.[24]

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[23] "Condemnation is the permanent relocation of the affected population if decontamination, natural weathering, and radioactive decay cannot adequately reduce contamination levels to habitability limits within 30 years" (USNRC, 2013, p. 103).

[24] Only one study (Benjamin et al., 1979) was available at the time the policy statement was developed. The first USNRC regulatory analysis dealing with spent fuel pools (Throm, 1989) was carried out after the policy statement was issued.

The USNRC staff recognized the limitations of the USNRC's screening goals for analyzing spent fuel pool accidents:

> "The QHOs [quantitative health objectives] effectively establish expectations related to the frequency of severe accidents associated with nuclear reactors and the potential for release of radioactive materials from an operating reactor core. . . . Some considerations in comparing SFP [spent fuel pool] risks to the QHOs are that the potential consequences of a SFP accident can exceed those of reactor accidents in terms of the amount of long-lived radioactive material released, the land area affected, and the economic consequences." (USNRC, 2013, p. 9)

In fact, a spent fuel pool accident can result in large radioactive material releases, extensive land contamination, and large-scale population dislocations. For example, Figures 7.3A-C show the estimated radioactive

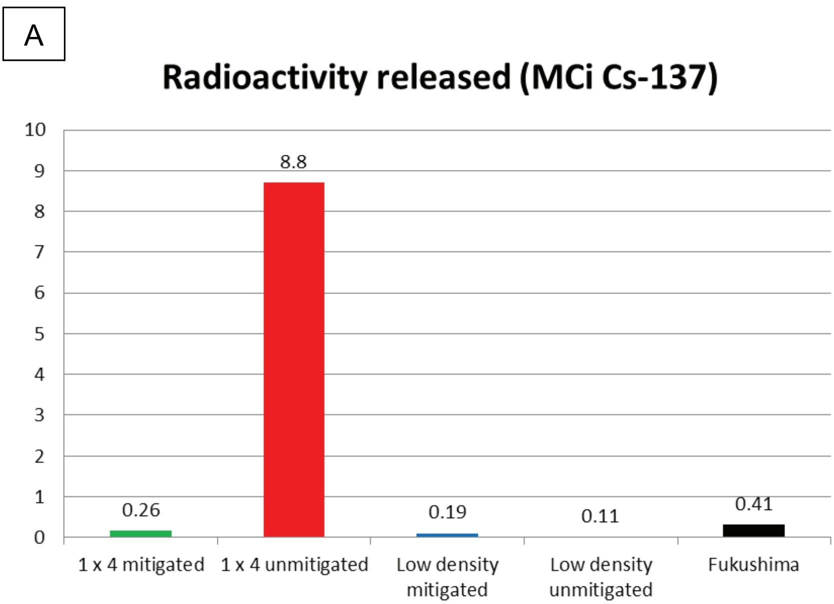
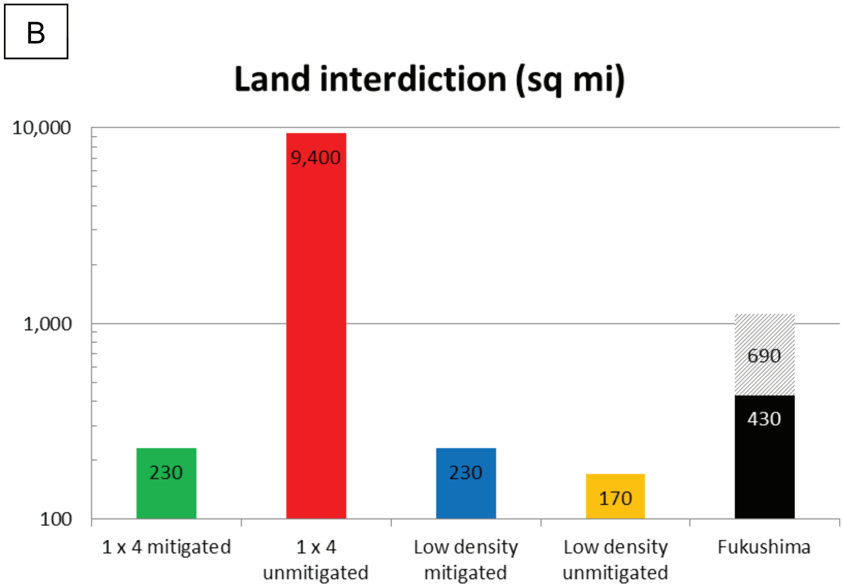


FIGURE 7.3 Selected consequences from the Spent Fuel Pool Study as a function of fuel loading (1 × 4 loading; low-density loading) and mitigation required by 10 CFR 50.54(hh)(2). Notes: Consequences for the Fukushima Daiichi accident are shown for comparison. (A) Radioactive material releases. (B) Land interdiction (see footnote 26 for an explanation of the values for the Fukushima bar). (C) Displaced populations. SOURCE: Table 7.1 in this report; IAEA (2015), NRA (2013), NRC (2014, Chapter 6), UNSCEAR (2013).



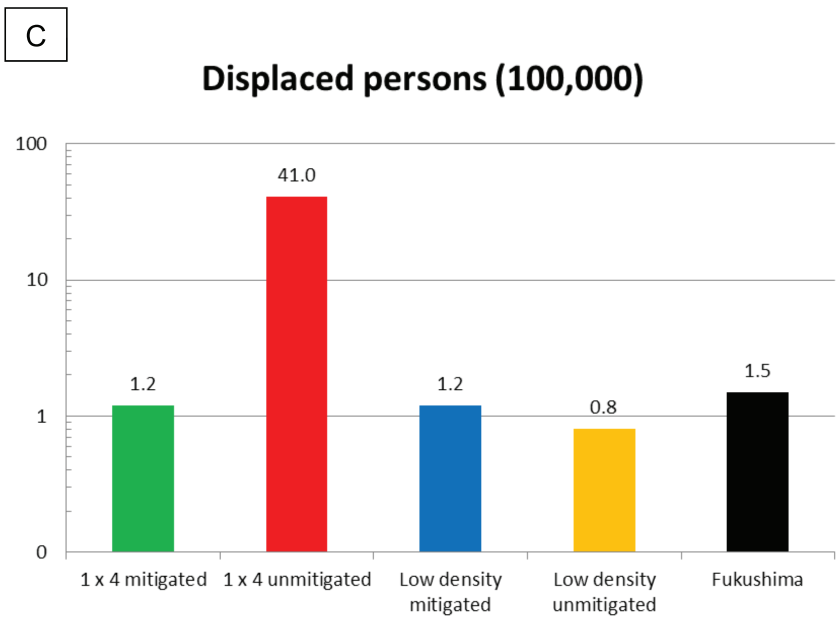


FIGURE 7.3 Continued

material releases, land interdiction, and displaced persons for the reference plant in the Spent Fuel Pool Study (see Table 7.1). Also shown for comparison purposes are the same consequences for the Fukushima Daiichi accident taken from the committee's phase 1 report (see NRC, 2014, Chapter 6).

These figures illustrate three important points:

1. A spent fuel pool accident can result in large releases of radioactive material, extensive land interdiction, and large population displacements.
2. Effective mitigation of such accidents can substantially reduce these consequences for some fuel configurations (cf. the bars in the figures for 1 × 4 mitigated and unmitigated scenarios) but can increase consequences for others (cf. the bars in the figures for low-density unmitigated and unmitigated scenarios).[25]
3. Low-density loading of spent fuel in pools can also substantially reduce these consequences and also reduce the need for effective mitigation measures.

The above points are not obvious when consequence estimates are presented only after being weighted by release frequencies. The committee judges that it is important to present the full risk triplet (scenarios, frequencies, and consequences) separately, as well as their product, in cost-benefit analyses.

Note that the Fukushima estimate includes land that is both interdicted[26] and likely condemned (see footnote 23 for the definition of condemned land); the Spent Fuel Study (USNRC, 2014a) reports only interdicted land. One of the difficulties with USNRC (2014a) is that, unlike previous studies, the condemned land is not reported. Of the 430 mi^2 (1,113 km^2) that were

[25] This increase in consequences is the result of larger water inventories in the pools from removal of spent fuel assemblies to attain low-density configurations. It takes longer to drain the pools below the baseplates at the bottoms of the fuel racks because the pool contains more water, which delays the establishment of natural air convection through the fuel assemblies to prevent them from reaching runaway-oxidation conditions.

[26] Interdiction is defined in footnote 12. Interdicted land is temporarily evacuated during the first year due to the dose level exceeding 500 mrem/year (5 mSv/yr); see footnote f to Table 7.1. We report two values for interdicted land as a result of the Fukushima accident: the mandated evacuation area of 430 mi^2 (1,113 km^2) in 2013 (NRA, 2013) and the additional area of 690 mi^2 (1787 km^2) contaminated to an excess dose level of 100 mrem/yr (1 mSv/yr) evaluated from the projected excess dose mapping by IAEA (2015). The interdicted land reported here is much smaller than that reported in NRC (2014, p. 6-4) because that value was based on an estimate made immediately after the accident and before detailed radioactivity surveys were available. The government of Japan has lifted evacuation orders in some regions that have been decontaminated to projected dose levels less than 20 mSv/yr (MOE, 2015, pp. 8-9) and aims to lift more orders as decontamination efforts warrant. In some instances, the projected dose levels in these areas are higher than the government's proposed 1 mSv/yr long-term cleanup target (see MOE, 2015, p. 28).

evacuated as of May 2013, 124 mi^2 (320 km^2) was reported as "difficult to return," which gives an indication of the amount of land that may ultimately be condemned.

A similar point can be made by examining the unweighted results from the Expedited Transfer Regulatory Analysis (USNRC, 2013) for a "sensitivity case" that removes the 50-mile limit for land interdiction and population displacements and raises the value of the averted dose conversion factor from $2,000 per person-rem to $4,000 per person-rem.[27] This scenario postulates the evacuation of 3.46 million people from an area of 11,920 mi^2, larger than the area of New Jersey (Table 7.2).[28]

In fact removing the 50-mile limit and raising the value of the averted dose conversion factor to $4,000 per person-rem increased the base-case average estimated benefits of expedited transfer by a factor of 5.9, that is, from about 13 percent of the estimated costs of expedited transfer to about 80 percent.[29] Moreover, for the 20 reactors with shared spent fuel pools and the four AP1000 reactors currently under construction (see Section 7.3), the base-case benefits were found to exceed the costs of expedited transfer (i.e., expedited transfer would have been cost beneficial), even though the base case had a limited safety benefit when assessed against the QHOs.

The numbers presented in Table 7.2 are not weighted by frequency. Consequently, they are not expected values and cannot be compared directly with the cost-benefit results in the Expedited Transfer Regulatory Analysis (USNRC, 2013).

The cost-benefit analysis did not consider some other important health consequences of spent fuel pool accidents, in particular social distress. The Fukushima Daiichi accident produced considerable psychological stresses within populations in the Fukushima Prefecture over the past 4 years, even in areas where radiation levels are deemed by regulators to be acceptable for habitation. Radiation anxiety, insomnia, and alcohol misuse were significantly elevated 3 years after the accident (Karz et al., 2014). The incidence of mental health problems and suicidal thoughts also were high among residents forced to live in long-term shelters after the accident (Amagai

[27] Current USNRC guidance (USNRC, 2004) specifies the use of $2,000 per person-rem for the averted dose conversion factor. The USNRC is in the process of revising this factor. The $4,000 per person-rem value used in the USNRC's sensitivity analysis is based on the updated Environmental Protection Agency value of a statistical life and the International Commission on Radiation Protection mortality risk factor coefficient (USNRC, 2013, p. 120). However, the value of the averted dose conversion factor is a matter of Commission policy.

[28] In comparison, approximately 88,000 people were involuntarily displaced from an area of about 400 mi^2 as a consequence of the Fukushima accident (MOE, 2015).

[29] These average estimated benefits were obtained using the Group 1-4 pool frequencies in Table 1 and the cost-benefit estimates in Tables 10, 27, 28, 29, and 30 for the 7 percent discount rate case in USNRC (2013).

TABLE 7.2 Sensitivity Scenario of Pool-Averaged Consequences and Benefits for Expedited Transfer

Cost or Benefit[a]	Sensitivity Study Base-Case Average (range)[b]
Area interdicted (mi^2)	11,900 (5,220-18,500)
Population interdicted (million)	3.46 (1.34-8.68)
Population dose cost ($billion)	435 (84-1,133)
Property loss ($billion)	265 (85-668)
Total benefits from expedited transfer ($billion)	701 (170-1,802)

NOTE: These results are averaged over the four spent fuel pool groups, weighted by the number of pools in each group, and have not been weighted by release frequencies.
[a] Costs and benefits are in 2012 dollars, and no discount factors have been applied. The changes in the costs of potential dry cask storage accidents are not included.
[b] This sensitivity case eliminates the 50-mile restriction on land contamination and population displacements and uses a $4,000 per person-rem averted dose conversion factor. The pool-weighted release of cesium-137 from a high-density pool accident for the base case in USNRC (2013) is estimated to be 43 MCi. This estimate was obtained by multiplying the cesium-137 inventories in high-density pools (Table 35) by the release fractions from high-density pools (Table 2) and weighting the results by the Group 1-4 pool frequencies in USNRC (2013).
SOURCE: USNRC, written communications, July 15, 2015, and March 8, 2016.

et al., 2014). Complex psychosocial effects were also observed, including discordance within families over perceptions of radiation risk, between families over unequal compensatory treatments, and between evacuees and their host communities (Hasegawa et al., 2015).

These findings are not new. Ten years after the 1979 TMI accident, for example, worries about personal and children's health were still elevated among women who had lived within 10 miles of the plant prior to the accident (Bromet and Licher-Kelly, 2002), despite the fact that radioactive releases from that accident were small.

Well-documented mental health impacts have also been seen in populations affected by the 1986 Chernobyl accident. Danzer and Danzer (2014) analyzed a sample of adults drawn from the population that was not relocated out of areas contaminated by the accident. They used survey and economic data to estimate the increase in national income that would be needed to compensate for the impact of the accident on life satisfaction: about 6 percent of Ukraine's gross domestic product. Masunaga et al. (2014) found that even well-educated people born after the Chernobyl accident in areas that were only modestly contaminated had anxiety about their radiation exposures, which has affected their mental health.

It is too soon to know what the long-term mental health impacts will be in the Japanese populations affected by the Fukushima Daiichi accident.

It is clear, however, that mental health impacts are a major if not dominant effect from nuclear accidents involving land contamination and have potentially large attendant costs. Many of these impacts are not readily monetizable at present but could be assessed qualitatively.

Additional research might be needed to develop quantitative metrics for social distress. The development of such metrics might at first glance seem daunting given the myriad ways social distress can be displayed in human populations. On the other hand, relatively simple metrics might be developed based on the underlying drivers for social distress, namely land contamination and population relocations. Such metrics could include, for example, a cost metric based on land areas contaminated above certain thresholds that would require temporary or permanent relocations or remedial actions, as well as a population metric based on the numbers, ages, and employment status of affected people.

7.4.4 Bounding Assumptions

The USNRC staff used numerous bounding assumptions in the Expedited Transfer Regulatory Analysis to "ensure that design, operational, and other site variations among the new and operating reactor fleet were addressed and to generally increase the calculated benefits from the proposed action" (USNRC, 2013, p. 7 in Memorandum to Commissioners). Bounding assumptions were used, for example, for

- Frequency of damage to spent fuel pools from accident initiators that could challenge pool cooling or integrity,
- Loss of AC power following an accident initiator,
- Potential drainage paths from pools,
- Potential for natural-circulation air cooling following drainage, and
- Conditional probability for the failure to successfully mitigate an accident.

Bounding assumptions are commonly used in safety assessments to account for variabilities in model parameters and unanalyzed risks. However, the use of such assumptions can make it difficult to determine whether the results of an analysis are truly bounding. Moreover, it can be difficult or impossible to assign confidence intervals[30] to the results when parameter uncertainties are not propagated through the analysis.

[30] A confidence interval expresses the degree of uncertainty associated with a model result. It is usually expressed in terms of a probability, for example, a 95 percent probability that the model result falls within the stated uncertainty range.

Sensitivity tests can be used to examine the effects of particular bounding assumptions on the results of an analysis. However, these tests are usually carried out by varying one parameter at a time while holding the other parameters at fixed values. This approach, which was used in the Expedited Transfer Regulatory Analysis, does not account for potential parameter covariability. This approach also makes it difficult to propagate parameter value uncertainties through the analysis to estimate uncertainties in the expected consequences.

It can be difficult to perform valid comparisons of analysis results without reliable uncertainty estimates. To illustrate, consider two spent fuel pool accident scenarios that yield similar best-estimate probability-weighted consequences. The first scenario involves a high-probability, low-consequence event that has a small uncertainty of occurrence. (The uncertainty is small because the event occurs frequently enough to be observed and measured.) The second scenario involves a low-probability, high-consequence event that has a large uncertainty of occurrence. (The uncertainty is large because the event occurs very infrequently and may not have been observed or measured.)

The best-estimate consequences for these two scenarios might have similar numerical values. However, their confidence levels are different—the high-probability event has a high confidence level (i.e., low uncertainty) compared to the low-probability event. Consequently, one would need to know both the best-estimate values and their uncertainty ranges to make useful risk comparisons.

Table 7.2 shows selected accident consequences and cost estimates for a base-case scenario for the Expedited Transfer Regulatory Analysis. Also shown are the ranges of low and high estimates from the analyses. It is immediately apparent that the ranges are large. When weighted by probability, these ranges overlap the cost estimates in the regulatory analysis. This example illustrates the limitations of using best estimates in isolation for making policy decisions.

7.4.5 Concluding Comments

The committee judges that the most effective means to assess the need for expedited transfer would be through a risk assessment that addresses the three questions of the risk triplet (see Chapter 5) and that accounts for uncertainties in both probability and consequence estimates. Such an assessment could include qualitative assessments of currently nonquantifiable consequences such as mental health impacts, or an effort could be made to quantify such impacts. The Spent Fuel Pool Study is "a limited-scope consequence assessment that utilizes probabilistic insights" (USNRC, 2014a, p. 6). It is not a risk assessment. The study is, however, a useful step toward a risk assessment of spent fuel storage arrangements.

The committee's recommended assessment of spent fuel storage risks would go beyond the Expedited Transfer Regulatory Analysis to include

1. Use of established methods to evaluate risk from accidents and sabotage in terms of the risk triplet: scenarios, likelihoods, and consequences. These methods are discussed in the committee's phase 1 report (NRC, 2014)—see especially Chapter 5 and Appendix I in that report—and in Chapter 5 of the present report. The committee anticipates that the accident and sabotage risk assessments would be carried out separately because they use different analytical approaches. However, there would likely be some commonalities in the event progression and consequence analysis portions of the two assessments.
2. The safety and sabotage risks for dry cask storage.
3. The range of expected economic and health consequences that would likely result from a severe nuclear accident, as seen most recently in Japan following the Fukushima Daiichi accident. Cost and health impacts associated with land interdiction and population relocation need to accurately reflect the implications of the Japanese experience for U.S. conditions.

The committee-recommended risk assessment would be particularly valuable for analyzing pool storage risks in plants that are in outage or undergoing decommissioning. During plant outages, the reactor core may be moved into the pool to facilitate refueling or maintenance, substantially increasing pool heat loads. During plant decommissioning, the pool may be filled to near capacity and some plant safety systems may be inoperable.

The committee discusses how the USNRC might carry out a sabotage risk assessment in Chapter 5. Although there remain differences of opinion regarding the extent to which risk assessment methods can be meaningfully applied to terrorist threat, clear progress is being made in developing and applying risk assessment methods to terrorist threat. Chapter 5 documents several examples of quantitative assessments of the risks associated with terrorist threats. There are important insights to be gained from more in-depth analysis of these risks, particularly the risks associated with insider cyber threats.

Risk analysis tools that focus on the risk triplet—scenarios, likelihoods, and consequences—can contribute to those insights. The numerical results of such analyses can be used to make relative comparisons, for example, to compare differences in design or operational alternatives within a particular system or facility or between facilities, particularly when the analyses are conducted by the same group of people applying comparable assumptions.

Even if the USNRC staff were to determine that substantially more thorough quantification of sabotage risks is not feasible at this time it could undertake qualitative or partially quantitative analyses. Whichever approach is used, the risk assessment should identify, communicate, and account for the uncertainties in the analyses.

The USNRC staff informed the committee that it is already thinking about how to expand its risk assessment methodologies to include sabotage risks. The committee strongly encourages the staff to continue this important effort.

The decision to expedite the transfer of spent fuel from pools to dry casks is a policy decision for the USNRC, not the task of this study. The committee's critiques of the Spent Fuel Pool Study and Expedited Transfer Regulatory Analysis are intended to strengthen the quality of any future analyses of spent fuel pool storage risks to support sound decision making by the USNRC and nuclear industry.

APPENDIX 7A
Spent Fuel Pool Racking

The Spent Fuel Pool Study (USNRC, 2014a) did not analyze the effects of "open" or "low-density" racking on the coolability of spent fuel in air. USNRC staff noted that

> "Re-racking the pool would represent a significant expense, along with additional worker dose, and was not felt to be the likely regulatory approach taken based on consultation with the Office of Nuclear Reactor Regulation. Much of the benefit of low density racking is achieved by the implementation of a favorable fuel pattern (1 × 4). Additionally, to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed." (USNRC, 2014a, p. 23)

And

> "Based on insights from the SFPS [Spent Fuel Pool Study], the [USNRC] staff believes that within the first few months after the fuel came out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of convective cooling. Therefore, reracking the SFP [Spent Fuel Pool] to install open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time." (USNRC, 2013, p. 31)

In response to a question from the public about whether the results of the Expedited Transfer Regulatory Analysis (USNRC, 2013) would change if open-frame racks were considered, USNRC staff noted the following:

> "For the reference plant studied, the BWR fuel assemblies channel boxes would impede crossflow even with open-frame racks. Furthermore, even for the high-density racking, the study showed that without mitigative actions, fuel is estimated to be air-coolable for at least 72 hours for all but roughly 10% of the operating cycle. Based on the insights from the accident progression analyses in the SFPS, within the first few months after the fuel comes out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of any additional convective cooling once natural circulation is established (see Figures 90 and 93 in the SFPS for the high-density and low-density pool loadings and a moderate leak). Therefore, open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time." (USNRC, 2013, p. 139)

The USNRC's concerns about the "significant expense" of reracking, the need to remove channel boxes to obtain an appreciable benefit in BWR pools, and increased worker exposures are plausible; however, there is no supporting analysis in USNRC (2013, 2014a) regarding the potential benefits of low-density racks. On the other hand, the argument against considering low-density racking has merit: Within a certain time period after the fuel is removed from a reactor, single isolated fuel assemblies cannot be safely cooled by natural air convection alone. This suggests that a limited benefit would be obtained by going to a low-density rack configuration.

Evaluating the efficacy of open racking requires modeling natural convective cooling of widely spaced assemblies in air, water, or multiphase mixtures, particularly under conditions where oxidation may take place. Flow between widely spaced fuel assemblies will be countercurrent and three-dimensional, driven by buoyancy differences between water or air masses in the pool. The control-volume approach of MELCOR, which was used in the Spent Fuel Pool Study (USNRC, 2014a), is poorly suited for modeling these types of flows. This model treats the large open portions of the pool and building as single volumes with well-defined mixing properties. One needs a computational fluid dynamics (CFD) model that solves the field equations for conservation of mass, momentum, and energy to properly represent the flows that would be expected to occur in pools with low-density racks (see Sidebar 6.1 in Chapter 6). This CFD model needs to be validated with experiments.

Flow in the pool with low-density racks will be turbulent. Given the characteristic dimensions of the pools and fuel racks, significant approximations of unknown fidelity will have to be used to model the fluid dynamics and heat transfer from the fuel rods to pool water. Even greater modeling difficulties will be encountered for partially drained pools because there will be two-phase three-component flow (liquid water, vapor water, air) within the open spaces of the fuel assemblies and a combination of vaporization and buoyancy-driven mixing between assemblies.

Sandia National Laboratories and the USNRC have carried out separate CFD studies on natural convection processes in fully drained pools; there have also been studies on water-filled pools by other researchers (e.g., Boyd, 2000; Chen et al., 2014; Hung et al., 2013; Wagner and Gauntt, 2008). Boyd (2000) discusses the limitations of natural convection for cooling spent fuel and uses a CFD model (FLUENT) to model air convection in a fully drained pool. None of these studies assessed the effects of fuel dispersal in the pool or open versus closed racking.

Benjamin et al. (1979) modeled a loss-of-coolant accident in a spent fuel pool with several rack configurations using a modeling approach similar to MELCOR. One of the configurations considered was an open frame that represented an early design spent fuel pool rack. They note that

> "The open frame configuration . . . is more difficult to analyze because of the lack of defined flow paths. On the other hand, it is obviously a very coolable configuration because of the openness of the structure and the large spacings between elements, so that a detailed exact flow calculation was not deemed necessary from a practical viewpoint." (p. 105)

Benjamin et al. used an abbreviated version of their model (SFUEL) to assess air circulation in an open frame rack, but they cautioned that

> "The calculations for the open frame configuration should be viewed as very approximate, with minimum allowable decay times being accurate, perhaps, to within a factor of two." (p. 106)

Sailor et al. (1987) used a modified version of SFUEL to estimate the risks (likelihoods) of zirconium cladding fires as a function of racking density. They estimated that risks could be reduced by a factor of 5 by switching from high- to low-density racks. This estimate was based on the reduction of minimum decay times before the fuel could be air cooled, and also on the reduction in the likelihood of propagation of a zirconium cladding fire from recently discharged fuel assemblies to older fuel assemblies in the low-density racks compared to high-density racks. However, Sailor et al. (1987) cautioned that "[t]he uncertainties in the risk estimate are large."

The regulatory analysis for the resolution of Generic Issue 82[1] (Throm, 1989) was intended to determine whether the use of high-density racks poses an unacceptable risk to the health and safety of the public. The analysis concluded that no regulatory action was needed; that is, the use of high-density storage racks posed an acceptable risk. The technical analysis was based on the studies of Benjamin et al. (1979) and Sailor et al. (1987) and used the factor-of-5 reduction in the likelihood (i.e., the conditional probability of a fire given a drained pool) of a zirconium cladding fire for switching to low-density racks from high-density racks. A cost-benefit analysis analogous to that employed in USNRC (2014a) found that the costs associated with reracking existing pools (and moving older fuel in the pool to dry storage to accommodate reracking) substantially exceeded the benefits in terms of population dose reductions.

The assumptions and methodology used in the regulatory analysis for Generic Issue 82 are similar to those used in USNRC (2014a): A seismic event is considered the most likely initiator of the accident and spent fuel pool damage frequency is taken to be about 2×10^{-6} events per reactor-year. Moreover, USNRC (2014a) reached essentially the same conclusions as the regulatory analysis for the resolution of Generic Issue 82 (Throm,

[1] This study was undertaken in response to the issues identified in the Benjamin et al. (1979) study.

1989). However, USNRC (2014a) took more credit for the operating cycle in reducing the risks of zirconium cladding fires.

A more pessimistic view on the uncertainties of modeling spent fuel pool loss-of-coolant accidents was expressed by Collins and Hubbard (2001):

> "In its thermal-hydraulic analysis . . . the staff concluded that it was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible. Heat removal is very sensitive to these additional constraints, which involve factors such as fuel assembly geometry and SFP rack configuration. However, fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis." (p. 5-2)

The older studies of Benjamin et al. (1979) and Sailor et al. (1987) simulated open racking configurations and showed the potential for increased air coolability for those configurations. More recent analyses by Sandia National Laboratories (see Chapter 6) and the Spent Fuel Pool Study (USNRC, 2014a) have not carried out simulations of these configurations. There have been substantial advances over the past decade in understanding the complex phenomena involved in the prediction of critical conditions for fuel assembly ignition. Consequently, these older studies will need to be revisited as part of any future consideration of reracking as a spent fuel pool management strategy. As discussed in this Appendix, the modeling approach and software (MELCOR) used in USNRC (2014a) have limitations that will need to be addressed as part of any study of reracking. Accurate modeling of natural convective cooling of widely spaced fuel assemblies will require careful examination of the fundamental assumptions in the modeling and validation against test data.

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Appendix A

Committee and Staff Biographies

Joseph E. Shepherd, Chair. Joseph E. Shepherd, Ph.D., is the C. L. "Kelly" Johnson Professor of Aeronautics and Mechanical Engineering and vice president for student affairs at the California Institute of Technology (Caltech) in Pasadena. His research interests are fluid and solid dynamics, combustion chemistry, thermodynamics, and dynamic structural response with applications to explosions, propulsion, high-speed flight, and energy technology. Since 1980 he has carried out research on hydrogen combustion behavior during severe accidents in nuclear power plants as well as in nuclear material processing and storage facilities. He has worked with private industry, the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, U.S. National Laboratories, and international organizations to evaluate hydrogen control methodologies and assess potential hazards including the effects of explosions. Dr. Shepherd earned his Ph.D. in applied physics from Caltech in 1981. He served as a member of the technical staff at Sandia National Laboratories from 1980 to 1986. From 1986 to 1993 Dr. Shepherd was an assistant professor of mechanical engineering at Rensselaer Polytechnic Institute. He has been on the faculty at Caltech since 1993. Dr. Shepherd served on the National Research Council Committee on Determining Basic Research Needs to Interrupt the Improvised Explosive Device Delivery Chain.

Robert A. Bari. Robert A. Bari, Ph.D., is a senior physicist at Brookhaven National Laboratory. He has more than 40 years of experience in the field of nuclear energy and has directed numerous studies of advanced nuclear energy concepts involving nuclear energy technology performance, nonproliferation, safety, economics, and waste management. For more than 25 years, Dr. Bari served at all levels of management at Brookhaven National Laboratory. He is currently international co-chairman of the working group of the Generation IV International Forum (Gen IV) that has developed a comprehensive methodology for evaluation of proliferation resistance and physical protection of all new nuclear energy concepts being proposed within by Gen IV. Dr. Bari has served on the Board of Directors of the American Nuclear Society (ANS) and is past chairman of the ANS Consensus Standards Committee for Probabilistic Risk Assessment. He is past president of the International Association for Probabilistic Safety Assessment and Management. For his achievements in nuclear safety, he was awarded the Theo J. "Tommy" Thompson Award in 2003 by the American Nuclear Society. In 2004, he received the Brookhaven National Laboratory Award for Outstanding Achievement in Science and Technology. He was awarded membership in the Phi Beta Kappa, Sigma Xi, and Sigma Pi Sigma honor societies and is an elected fellow of both the American Nuclear Society and the American Physical Society. Dr. Bari has served as an adjunct faculty member and advisor to several major universities in the field of nuclear technology. Dr. Bari recently chaired a workshop of the U.S. National Academy of Sciences on safety and security culture held jointly between the United States and Brazil in Sao Paolo. He received his doctorate in physics from Brandeis University (1970) and his bachelor's degree in physics from Rutgers University (1965).

Jan Beyea. Jan Beyea, Ph.D., is chief scientist at Consulting in the Public Interest, where he consults on energy/environmental topics for local, national, and international organizations. He has expertise in energy technologies and associated environmental and health concerns and has written numerous articles on energy and the environment, including articles on planning for reactor accidents. His current research interests are in the field of epidemiology. Dr. Beyea previously served as chief scientist and vice president of the National Audubon Society and held positions at Holy Cross College, Columbia University, and Princeton University's Center for Energy and Environmental Studies. He received a B.A. from Amherst College and a Ph.D. in physics from Columbia University. Dr. Beyea has been a member of numerous advisory committees and panels, including the National Research Council's Board on Energy and Environmental Systems, Energy Engineering Board, Committee on Assessment of the Prospects for Inertial Fusion Energy, Committee on America's Energy Future, Committee on Alternative Energy R&D Strategies, Committee to Review DOE's Fine Particulates Research Plan, and Committee on Alternatives for Controlling the Release of Solid Materials from Nuclear Regulatory Commission-Licensed Facilities. He has also served on the Secretary of Energy Advisory Board's Task

Force on Economic Modeling, has been a member of the policy committee of the Recycling Advisory Council, and has advised various studies of the Office of Technology Assessment. In 2012, he recently served as a guest editor for, and contributor to, a theme issue of the *Bulletin of the Atomic Scientists* on the subject of risks from low-level radiation. Dr. Beyea is an elected Fellow of the American Physical Society.

Michael L. Corradini. Michael L. Corradini, Ph.D., is a professor in the Department of Engineering Physics at the University of Wisconsin, Madison. Dr. Corradini's research focus is nuclear engineering and multiphase flow with specific interests that include light-water reactor safety, fusion reactor design and safety, waste management and disposal, vapor explosions research and molten core–concrete interaction research, and energy policy analysis. He received his B.S. in mechanical engineering from Marquette University and his M.S. and Ph.D. degrees in nuclear engineering from the Massachusetts Institute of Technology. He is a member of the American Institute of Chemical Engineers, the American Society of Engineering Education, and the American Society of Mechanical Engineers, and a Fellow of the American Nuclear Society. Dr. Corradini has received numerous awards including the National Science Foundation's Presidential Young Investigator Award, the American Nuclear Society reactor safety best paper award, and the University of Wisconsin, Madison, campus teaching award. He has served on various technical review committees, including the research review panel of the U.S. Nuclear Regulatory Commission. He currently serves on the U.S. Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards and was president of the American Nuclear Society in 2013-2014. Dr. Corradini was elected to the National Academy of Engineering in 1998.

Vijay K. Dhir. Vijay K. Dhir, Ph.D., is a distinguished professor of mechanical and aerospace engineering and was dean of the University of California, Los Angeles (UCLA) Henry Samueli School of Engineering and Applied Science from 2003 to 2015. He also leads the boiling heat transfer laboratory, which conducts pioneering work in fundamental and applied sciences involving boiling, an efficient process of heat removal. Currently, his laboratory is involved in the study of flow boiling, microgravity boiling, and nuclear reactor thermal hydraulics. Born in India, Dr. Dhir received his B.Sc. from Punjab Engineering College in Chandigarh, India, and his master of technology from the Indian Institute of Technology in Kanpur, India. He received his Ph.D. from the University of Kentucky. In the late 1960s, he worked for a short period in industry as an engineer, and for the past 35 years he has been a consultant for numerous organizations. Dr. Dhir served as chair of the UCLA Department of Mechanical and

Aerospace Engineering from 1994 to 2000. In 2004, he was selected as an inductee into the University of Kentucky's Engineering Hall of Distinction, and in 2012 he received his alma mater's honorary Ph.D. degree. The American Society of Mechanical Engineers (ASME) has honored him with the Heat Transfer Memorial Award and the Robert Henry Thurston Lecture Award. The American Institute of Chemical Engineers honored him with the Donald Q. Kern award and the Max Jakob Memorial Award (awarded jointly with ASME). He is recipient of the Technical Achievement Award of the Thermal Hydraulics Division of the American Nuclear Society. Most recently, he received the Lifetime Achievement Award at ICCES (the International Conference on Computational & Experimental Engineering and Sciences). Dr. Dhir has more than 300 publications in archival journals and proceedings of conferences. He was elected to the National Academy of Engineering in 2006.

Michael W. Golay. Michael W. Golay, Ph.D., is a professor of nuclear science and engineering at the Massachusetts Institute of Technology (MIT), where he has worked since 1971. He is director of the Reactor Technology Course for Utility Executives and the Nuclear Operational Risk Management Course, both cosponsored by MIT and the National Academy for Nuclear Training. Most recently he has focused his research and teaching on improving nuclear power performance both in the United States and internationally, particularly through use of probabilistic and dynamic methods of analysis. He has also been an active advisor to governmental and industrial organizations, particularly concerning risk-informed regulation and nuclear nonproliferation. Dr. Golay received his Ph.D. in nuclear engineering from Cornell University in 1969 and performed postdoctoral research at Rennselaer Polytechnic Institute. In 1980, he was a visiting researcher at Électricité de France. He has served on the Institute of Nuclear Power Operations Advisory Council, the U.S. Nuclear Regulatory Commission's Research Review Committee, the Department of Energy's TOPS Committee (on nonproliferation), and national laboratory and nuclear power plant oversight committees. He is a fellow of the American Association for the Advancement of Science and the American Nuclear Society.

Barbara L. Hamrick. Barbara L. Hamrick, J.D., CHP, is the radiation safety officer at the University of California, Irvine, Medical Center where she oversees the use of radiation and radioactive materials in medical and research applications, including use in radiology, nuclear medicine, radiation oncology, pathology, and neurology. Ms. Hamrick received a B.S. and an M.S. in physics from the University of California, Irvine, in 1985 and 1987, respectively. She also received a J.D. from Loyola Law School in Los Angeles and was admitted to the California State Bar in

1999. Prior to joining the staff at the Medical Center, Ms. Hamrick worked for 18 years in regulatory agencies at the local, state, and federal levels, including the Los Angeles County Office of Radiation Management, the California Department of Public Health, and the U.S. Nuclear Regulatory Commission. Her work involves the application of health physics to a diverse set of problems, including survey and remediation at decommissioning facilities, external and internal dose assessments, air and water effluent modeling and monitoring, emergency planning, population monitoring, and radioactive waste management and disposal. Ms. Hamrick also has extensive experience and knowledge related to federal and state statutes and regulations governing the use of radiation and radioactive materials. She has participated in and co-chaired interagency working groups established to develop regulation, policy, and guidance related to the use of radiation and radioactive material in coordination with the U.S. Nuclear Regulatory Commission, U.S. Department of Homeland Security, U.S. Department of Energy, U.S. Environmental Protection Agency, Organization of Agreement States, and Conference of Radiation Control Program Directors. Ms. Hamrick is currently serving as a member of the U.S. Environmental Protection Agency Radiation Advisory Committee. She also serves as past president of the Health Physics Society. She was certified by the American Board of Health Physics in 2002.

Paul A. Locke. Paul A. Locke, Dr.P.H., J.D., M.P.H., an environmental health scientist and attorney, is an associate professor at the Johns Hopkins University Bloomberg School of Public Health. He holds his primary appointment in the Department of Environmental Health Sciences and a joint appointment in the Department of Health Policy and Management. Dr. Locke directs the Doctor of Public Health Program in Environmental Health Sciences. Dr. Locke's research and practice focus on how decision makers use scientific data and research in regulation and policy making and how environmental health sciences influence the policy-making process. His areas of study include designing and evaluating radiation protection initiatives and radiation policies, especially in low-dose radiation science, radon risk reduction, safe disposal of high-level radioactive waste, and uranium mining and recovery operations. He holds an M.P.H. from Yale University School of Medicine, a Dr.P.H. from the Johns Hopkins University Bloomberg School of Public Health, and a J.D. from Vanderbilt University School of Law. Dr. Locke was a member of the National Academy of Sciences Nuclear and Radiation Studies Board from 2003 to 2009, and chaired the National Academy's Committee on Uranium Mining in Virginia. Dr. Locke is admitted to practice law before the bars of New York and the District of Columbia, the Southern District Court of New York, and the U.S. Supreme Court.

Thomas G. Moser. Thomas G. Moser, U.S. Navy (retired), is an independent consultant who provides antiterrorism and security expertise to federal, state, and local government entities and private-sector clients. He holds a B.S. in business administration from Waynesburg College in Pennsylvania and an M.B.A. from Southern Illinois University. While on active service as a career Navy SEAL Officer, Mr. Moser served as commanding officer of the Navy's unique antiterrorism RED CELL team, commanding officer of SEAL Team FOUR, commanding officer of the Naval Special Warfare Development Group (a classified special SEAL Unit), and as chief of staff at the Joint Special Operations Command. Following his naval career, Mr. Moser served as a counterterrorism and special operations consultant and exercise planner for Department of Defense Special Operations Units. He developed plans to exercise the nation's military and first-responder units' response to incidents involving the use of chemical, biological, radiological, and nuclear weapons of mass destruction. He later worked with the Department of Energy (DOE) as the site manager of the Andrews Air Force Base laboratory facility that was responsible for one of the nation's Nuclear Emergency Search Teams. Following service at DOE, Mr. Moser was selected to serve as one of the first Department of Homeland Security (DHS) Protective Security Advisors (PSAs) and served as PSA to the State of South Carolina, representing DHS as an onsite critical infrastructure and vulnerability assessment specialist. Mr. Moser participated in comprehensive security assessments at nuclear power plants and material production facilities in North and South Carolina and served on the National Research Council Committee on Risk-Based Approaches for Securing the DOE Nuclear Weapons Complex. More recently, Mr. Moser participated in a survey and assessment of the Coast Guard Service of a Gulf Coast Cooperative member state, addressing counterpiracy and countersmuggling missions in the Persian Gulf and Arabian Sea. Mr. Moser is an American Society of Industrial Security Certified Protection Professional and Physical Security Professional.

Arthur T. Motta. Arthur T. Motta, Ph.D., is Chair of the Nuclear Engineering Program and a professor of nuclear engineering and of materials science and engineering at Pennsylvania State University. His research focuses on the environmental degradation of materials in the reactor environment with specific emphasis on nuclear fuel cladding. His research interests include radiation damage, corrosion and hydrogen ingress, mechanical behavior of materials, and materials characterization. He holds a B.Sc. in mechanical engineering and an M.Sc. in nuclear engineering from the Federal University of Rio de Janeiro, Brazil, and a Ph.D. in nuclear engineering from the University of California, Berkeley. Before coming to Penn State, he worked for the CEA at the Centre for Nuclear Studies in Grenoble, France, and for Atomic Energy of Canada Limited at the Chalk River Laboratories in Canada. He is a member of the editorial board of the *Journal of Nuclear Materials*. He received the Mishima Award from the American Nuclear Society for sustained contributions to nuclear fuel research and has just been awarded the Kroll Medal from ASTM for siginificant contributions to zirconium metallurgy.

John A. Orcutt. John A. Orcutt, Ph.D., is a distinguished professor of geophysics at Scripps Institution of Oceanography and a secretary of the Navy/Chief of Naval Operations Oceanography Chair. He received his B.S. in mathematics and physics from the U.S. Naval Academy in Annapolis, his M.Sc. in physical chemistry as a Fulbright Scholar at the University of Liverpool, and his Ph.D. in earth sciences from the Scripps Institution of Oceanography. He served as a submariner in the U.S. Navy and was the Chief Engineer on USS Kamehameha including a shipyard overhaul with refueling of the nuclear plant. His research interests include the exploitation of information technology for the collection and processing of real-time environmental data as well as marine and continental seismology and geophysics. He is chair of the MEDEA Ocean Panel and recently completed a review of hydroacoustics monitoring by the UN Comprehensive Test Ban Treaty Office in the Indian Ocean. He is a charter member of the National Research Council Ocean Studies Board and is serving another two terms nearly 25 years after his first. He is the principal investigator of a BP research institute at Scripps, which began operations in 2004. He received the Ewing Medal from the U.S. Navy and the American Geophysical Union in 1994; the Newcomb-Cleveland Prize from the American Association for the Advancement of Science in 1983; and the Marine Technology Society's Lockheed Martin Award for Ocean Science and Technology in 2007. He chaired the National Research Council review of the National Oceanic and Atmospheric Administration (NOAA) Tsunami Warning System and the Ocean Panel of the Climate, Energy and National Security Committee. He served as president of the American Geophysical Union from 2004 to 2006 and was elected as an Honorary Fellow of the Royal Astronomical Society in 2005. He was elected to the American Philosophical Society in 2002 and the National Academy of Engineering in 2011.

Emilie M. Roth. Emilie M. Roth, Ph.D., is the owner and principal scientist of Roth Cognitive Engineering. A cognitive psychologist, Dr. Roth's work involves the analysis of human problem solving and decision making in real-world environments (e.g., military command and control, intelligence analysis, nuclear power plant emergencies, railroad operations, and surgery), and the impacts of support systems (e.g., computerized procedures, alarm systems, advanced graphical displays, and new forms of automation) on cognitive performance. Dr. Roth has conducted empirical studies of naturalistic decision making, developed and applied cognitive task analysis and cognitive work analysis techniques for understanding the cognitive demands imposed by work environments, and developed principles for effective decision support for individuals and teams. Dr. Roth has supported design of first-of-a-kind systems including the command center for a next-generation Navy ship; a next-generation nuclear power plant control room; and work-centered support systems for flight planning and monitoring for an Air Force organization. She received her Ph.D. in cognitive psychology from the University of Illinois at Urbana-Champaign. She serves on the editorial board of the journals *Human Factors* and the *Journal of Cognitive Engineering and Decision Making*. She was elected a Fellow of the Human Factors and Ergonomics Society. She recently participated in the National Research Council Committee on Human-System Design Support for Changing Technology.

Elizabeth Q. Ten Eyck. Elizabeth Q. Ten Eyck is president of ETE Consulting, Inc. She is an expert in domestic and international nuclear safeguards and security for government-owned and -licensed commercial nuclear facilities and has participated in vulnerability assessments of U.S. critical infrastructure for the Department of Homeland Security. Ms. Ten Eyck received her B.S. in electrical engineering from the University of Maryland. She has more than 30 years of career federal service, first as a security engineer for the U.S. Secret Service, then as director of the Office of Safeguards and Security for the U.S. Department of Energy, and, until she retired in 2000, as director of the Division of Fuel Cycle Safety and Safeguards for the U.S. Nuclear Regulatory Commission (USNRC), where she managed the safety and safeguards regulatory program for commercial fuel-cycle facilities. During her career at the USNRC, she also managed transportation activities and the safeguards program for nuclear power reactors. Ms. Ten Eyck served on the National Research Council Committee on Transportation of Radioactive Waste.

Frank N. von Hippel. Frank N. von Hippel, Ph.D., is a senior research physicist and professor of public and international affairs emeritus at Princeton University's Program on Science and Global Security, which he co-founded. In 1989, he co-founded the journal *Science & Global Security*. He was a founding co-chair and continues as a member of the nongovernmental International Panel on Fissile Materials, which includes experts from 17 countries and develops proposals for initiatives to reduce global stocks of plutonium and highly enriched uranium and the numbers of locations where they can be found. He received a Ph.D. in nuclear physics from the University of Oxford and a B.A. from Massachusetts Institute of Technology. As a former assistant director for national security in the White House Office of Science and Technology, Dr. von Hippel's areas of policy research include nuclear arms control and nonproliferation, energy, and checks and balances in policy making for technology. He has been involved in reactor safety issues since he served as a member of he American Physical Society's 1974-1975 Study Group on Light Water Reactor Safety. Prior to going to Princeton, he worked for 10 years in the field of elementary-particle theoretical physics. Dr. von Hippel's awards include the American Physical Society (APS) 2010 Leo Szilard Lectureship Award for outstanding work and leadership in using physics to illuminate public policy on nuclear arms control and nonproliferation, nuclear energy, and energy efficiency; the American Association for the Advancement of Science 1994 Hilliard Roderick Prize for Excellence in Science, Arms Control and International Security; a MacArthur Foundation Prize Fellowship (1993-1998); and the 1977 APS Forum Award for Promoting the Understanding of the Relationship of Physics and Society. Dr. von Hippel recently served on the National Research Council Committee on Best Practices for Nuclear Materials Protection, Control and Accounting.

Loring A. Wyllie, Jr. Loring A. Wyllie, Jr., M.S., is chairman emeritus of the board and senior principal at Degenkolb Engineers. He has more than 45 years of professional experience in seismic evaluations, analysis, and design of strengthening measures for improved seismic performance. He serves as consultant to several University of California campuses, various commercial and architectural clients, Department of Energy laboratories, and many others. He received his B.S. and M.S. from the University of California, Berkeley. He is a past chairman of the state historical building safety board, whose mandate is to evaluate and analyze methods for strengthening buildings that preserve their historic character. He is also the past president of the Earthquake Engineering Research Institute. His contributions to the profession of structural engineering were recognized by his election to the National Academy of Engineering in 1990. In 2007, he was honored with the prestigious Outstanding Projects and Leaders Lifetime Achievement Award by the American Society of Civil Engineers (ASCE). He was made an honorary member of the Structural Engineers Association of Northern California and Earthquake Engineering Research Institute. In recognition of his expertise in concrete design and performance, the American Concrete Institute named him an honorary member in 2000. Mr. Wyllie was elected an honorary member of ASCE in 2001.

Study Director

Kevin D. Crowley. Kevin D. Crowley, Ph.D., is senior board director of the Nuclear and Radiation Studies Board (NRSB) at the National Academies of Sciences, Engineering, and Medicine in Washington, DC. He is responsible for planning and managing the NRSB's portfolio of studies on radiation health effects, radioactive waste management and environmental cleanup, and nuclear security and terrorism and has personally directed or co-directed more than 25 Academies studies in these and other subject areas. Dr. Crowley also is the principal investigator of the Academies' Radiation Effects Research Foundation project, which provides scientific support for the long-term study of health effects arising from exposures to ionizing radiation among World War II atomic-bombing survivors. Dr. Crowley held positions at Miami University of Ohio, the University of Oklahoma, and the U.S. Geological Survey before joining the Academies staff in 1993. He received his M.A. and Ph.D. degrees in geology from Princeton University.

Appendix B

Presentations

WASHINGTON, DC, JANUARY 29, 2015

- Overview of spent fuel safety and security, including U.S. Nuclear Regulatory Commission (USNRC) actions in response to 2004 NAS report. Jennifer Uhle, Deputy Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- Spent fuel pool safety studies and analyses. Steven Jones, Senior Reactor Systems Engineer, Balance of Plant Branch, Division of Safety Systems, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- USNRC implementation of lessons learned from the Fukushima Dai-ichi accident. Eric Bowman, Special Advisor for Technical, Japan Lessons-Learned Division, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- Security of spent fuel. Sandra Wastler, Branch Chief, Materials and Waste Security Branch, Division of Security Policy, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission
- Safety of high-burnup spent fuel. Paul Clifford, Senior Technical Advisor for Reactor Fuel, Division of Safety Systems, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- Safety and security of spent fuel storage in the United States. Edwin Lyman, Senior Scientist, Global Security Program, Union of Concerned Scientists

- Actions taken by industry to improve spent fuel storage safety and security following the 2004 NAS report and the Fukushima Daiichi accident. Steven Kraft, Senor Technical Advisor, Nuclear Energy Institute; Phil Amway, Senior Staff Engineer, Severe Accident Management, Exelon
- Spent fuel and spent fuel storage facilities at Fukushima Daiichi. Kenji Tateiwa, Manager, Nuclear Power Programs, Washington, DC Office, Tokyo Electric Power Company

WASHINGTON, DC, MARCH 23, 2015

- Improvements to spent fuel pools at French nuclear power plants. Franck Bigot, Deputy Manager, Institut de Radioprotection et de Sûreté Nucléaire; Laurent Gilloteau, Deputy Head of Department, Nuclear Safety Division, PWR Safety Assessment Department, Nuclear Safety Division, Institut de Radioprotection et de Sûreté Nucléaire; Trégourès Nicolas, Scientific Officer of the DENOPI Project, Nuclear Safety Division, Safety Research, Institut de Radioprotection et de Sûreté Nucléaire; Axelle Portier, Nuclear Safety Engineer, Nuclear Safety Division, PWR Safety Assessment Department, Periodic Safety Reviews Section, Institut de Radioprotection et de Sûreté Nucléaire
- French PWR spent fuel pool safety. Axelle Portier, Nuclear Safety Engineer, Nuclear Safety Division, PWR Safety Assessment Department, Periodic Safety Reviews Section, Institut de Radioprotection et de Sûreté Nucléaire
- Needs in research and development. Trégourès Nicolas, Scientific Officer of the DENOPI Project, Nuclear Safety Division, Safety Research, Institut de Radioprotection et de Sûreté Nucléaire
- Research and development: DENOPI project. Trégourès Nicolas, Scientific Officer of the DENOPI Project, Nuclear Safety Division, Safety Research, Institut de Radioprotection et de Sûreté Nucléaire
- Nuclear Energy Institute perspectives on: Role of equipment and procedures required under 10 CFR 50.54(hh)(2) (formerly B5b) and how they differ from the FLEX equipment and procedures. Nick Pappas, Shift Manager, Arizona Public Service
- Nuclear Energy Institute perspectives on: Industry needs for 60 days to disperse spent fuel in pools after offloading from reactors; current industry practices (e.g., approaches and timing) for dispersing spent fuel in pools after offloading, including differences in practices at BWR and PWR plants. Kristopher Cummings, Senior Project Manager, Nuclear Energy Institute

- Nuclear Energy Institute perspectives on: Adequacy of security-related information sharing with industry by the U.S. Nuclear Regulatory Commission (discussion). David Kline, Director, Security, Nuclear Energy Institute; Kristopher Cummings, Senior Project Manager, Nuclear Energy Institute

WASHINGTON, DC, MARCH 23, 2015[1]

- Regulating spent fuel security: Introduction. Jennifer Uhle, Deputy Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- Overview of spent fuel security requirements. Ralph Way, Senior Level Advisor for Security, Division of Security Operations, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission
- Material control and accounting inspections. Glenn Tuttle, Material Control and Accountability Inspector, Material Control and Accountability Branch, Division of Fuel Cycle Safety, Safeguards and Environmental Review, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission
- Studies on terrorist attack scenarios: Structural analysis. Jose Pires, Senior Technical Advisor for Civil/Structural Engineering, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission
- Studies on terrorist attack scenarios: Accident progression analysis. Donald Helton, Senior Reliability and Risk Engineer, Probabilistic Risk Assessment Branch, Division of Risk Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission
- B.5.b and EA-12-049 (FLEX) strategies for spent fuel pools. Eric Bowman, Senior Advisor for Technical, Japan Lessons-Learned Division, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission
- Overview of security requirements for dry cask storage. Susan Stuchell, Senior Security Specialist, Materials and Waste Security Branch, Division of Security Policy, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission

[1] These presentations were made in an *information-gathering session not open to the public* because they contained classified, Safeguards, and other security-restricted information that is exempt from public release under the Freedom of Information Act.

MAY 8, 2015

- Risk informing security. Christiana Lui, Division Director, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission

Appendix C

Conversions and Units

Distance-Related Conversions		
kilometers (km) and miles (mi)	1 km = 0.62 mi	1 mi = 1.6 km
km^2 (square kilometers) and mi^2 (square miles)	1 km^2 = 0.39 mi^2	1 mi^2 = 2.59 km^2
m (meters) and ft (feet)	1 m = 3.28 ft	1 ft = 0.30 m
m^3 (cubic meters) and ft^3 (cubic feet)	1 m^3 = 35.32 ft^3	1 ft^3 = 0.03 m^3
km/hr (kilometers per hour) and mph (miles per hour)	1 km/hr = 0.62 mph	1 mph = 1.6 km/hr
Radiation-Related Conversions		
mSv (millisieverts), mrem (millirem), and mGy[a] (milligray)	1 mSv = 100 mrem = 1 mGy	1 mrem = 0.01 mSv = 0.01 mGy
Bq (becquerels) and Ci (curies)	1 Bq = 2.7×10^{-11} Ci	1 Ci = 3.7×10^{10} Bq
Other		
MJ (megajoules) and kWhr (kilowatt hours)	1 MJ = 0.28 kWhr	1 kWhr = 3.6 MJ
MPa (megapascals) and psi (pounds per square inch)	1 MPa = 145 psi	1 psi = 0.007 MPa
Celsius and Fahrenheit	°C = (5/9)* (°F – 32°)	°F = (9/5)* °C +32°
metric tons and pounds (lbs)	1 metric ton = 2204.6 lbs	

Prefixes	
pico-	10^{-12}
micro-	10^{-6}
milli-	10^{-3}
kilo-	10^{3}
mega-	10^{6}
giga-	10^{9}
tera-	10^{12}
peta-	10^{15}

[a] Millisieverts and millirem are units of effective dose, whereas milligray is a unit of absorbed dose. They are numerically equivalent when exposure is from gamma rays and x-rays.

Unit of Measure	Abbreviation	Type of Measure
Becquerel	Bq	radiation activity
Celsius	C	temperature
centimeters	cm	distance
Fahrenheit	F	temperature
feet	ft	distance
gallon	gal	volume
gallons per minute	gpm	flow rate
Gray	Gy	absorbed radiation dose
Joule	J	energy
kilogram	kg	mass
kilometers	km	distance
kilopascals	kPa	pressure
kilovolts	kV	electrical potential
kilowatt	kW	electrical power
kilowatt-hour	kWhr	energy
liters per minute	Lpm	flow rate
megapascals	MPa	pressure
megawatts electric	MWe	electrical power
meters	m	distance
millimeters	mm	distance
millirem	mrem	effective radiation dose
millisievert	mSv	effective radiation dose
newton	N	force
pound	lb	mass
pounds per square inch	psi	pressure
volts	V	electrical potential

Appendix D

Acronyms

$	Dollar
¥	Yen
1D	One-dimensional
2D	Two-dimensional
3D	Three-dimensional
A	Area
A_0	Flow-passage area
AC	Alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AFM	Army Field Manual
ANS	American Nuclear Society
AP (reactor)	Advanced passive
ASME	American Society of Mechanical Engineers
B.5.b	Section B.5.b of Order EA-02-026
BAF	Bottom of active fuel
BNL	Brookhaven National Laboratory
BWR	Boiling water reactor
°C	Degrees Celsius
CARVER	Criticality + Accessibility + Recuperability + Vulnerability + Effect + Recognizability

C_f	Pressure drop coefficient
CFD	Computational fluid dynamics
CFR	Code of Federal Regulations
CI/KR	Critical Infrastructure and Key Resources
Cm	Centimeter
CNN	Cable News Network
COMSECY	Commission Action Memoranda, the Office of the Secretary (SECY)
C_P	Water specific heat capacity
CRS	Congressional Research Service
Cs	Cesium
D	Mass diffusivity of water vapor in air
DBA	Design-basis accident
DBT	Design-basis threat
DC	Direct current
DCLG	Department for Communities and Local Government (United Kingdom)
DHS	U.S. Department of Homeland Security
Diet	National Diet of Japan
DOE	U.S. Department of Energy
DPR	Decommissioning Planning Rule
DS	Dryer-separator
EA	Enforcement Action
EPRI	Electric Power Research Institute
°F	Degrees Fahrenheit
FLEX	Diverse and Flexible Coping Strategies
FOIA	Freedom of Information Act
FPC	Fuel Pool Cooling and Cleanup
ft	Feet
g	Gravity
GAO	U.S. Government Accountability Office
GE	General Electric
gpm	Gallons per minute
GTD	Global Terrorism Database
H	Hydrogen
ΔH_{fg}	Heat of vaporization
h_m	Mass transfer coefficient
hr	Hour

HRA	Human reliability analysis
IAEA	International Atomic Energy Agency
INMM	Institute of Nuclear Materials Management
ISFSI	Independent spent fuel storage installation
ISG	Interim Staff Guidance
J	Joules
JLD	Japan Lessons-Learned Division
K	Kelvin
kg	Kilogram
kJ	Kilojoule
km	Kilometer
L	Length
LCF	Latent cancer fatality
LOCA	Loss-of-coolant accident
m	Meter
M	Mass
MACCS2	MELCOR Accident Consequence Code System, version 2
MCi	Megacuries
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
mi	Miles
MJ	Megajoules
mm	Millimeter
MOE	Japan Ministry of the Environment
MOX	Mixed oxide
mrem	Millirem
mSv	Millisievert
MUWC	Make-Up Water Condensate
MW	Megawatt
NAIIC	Nuclear Accident Independent Investigation Commission (Japan)
NAP	National Academies Press
NARAC	National Atmospheric Release Advisory Center
NAS	National Academy of Sciences (United States), now the National Academies of Sciences, Engineering, and Medicine

NEA	Nuclear Energy Agency (Organisation for Economic Co-operation and Development)
NEI	Nuclear Energy Institute
NIPP	National Infrastructure Protection Plan
NISA	Nuclear and Industrial Safety Agency (Japan)
NRA	(Japan) Nuclear Regulation Authority
NRC	National Research Council
NSIR	Nuclear Security and Incident Response
NTTF	Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (U.S. Nuclear Regulatory Commission)
NUREG	(US)NRC technical report designation
O	Oxygen
OCP	Operating-cycle phase
OE	Office of Enforcement (USNRC)
OECD	Organisation for Economic Co-Operation and Development
OMB	U.S. Office of Management and Budget
ORNL	Oak Ridge National Laboratory
P	Pressure
P	Probability
PCT	Peak cladding temperature
PGA	Peak ground acceleration
PL	Public Law
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
$\dot{Q}$	Thermal power
QHO	Quantitative health objective
ρ	Fluid density
RAMCAP	Risk Assessment and Management for Critical Asset Protection
rem	Roentgen equivalent man
RISR	Risk Informed Security Regulation
RJIF	Rebuild Japan Initiative Foundation
RPG	Rocket-propelled grenade
s	Second
SAMG	Severe accident management guideline
SECY	Secretary of the Commission

SFP	Spent fuel pool
SFPS	Spent fuel pool study
Sh	Sherwood number
SNL	Sandia National Laboratories
SNM	Special nuclear materials
SOARCA	State-of-the-Art Reactor Consequence Analyses
SSP	Sector-Specific Plan
Sv	Sievert
t	Time
T	Temperature
TAF	Top of active fuel
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
U	Uranium
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USNRC	U.S. Nuclear Regulatory Commission
W	Width
WWII	World War II
yr	Year
Zr	Zirconium