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United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-1650
Revision 5

The United States of America Sixth National Report for the Convention on Nuclear Safety

Office of Nuclear Reactor Regulation

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The United States of America Sixth National Report for the Convention on Nuclear Safety

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Office of Nuclear Reactor Regulation

ABSTRACT

The United States (U.S.) Nuclear Regulatory Commission (NRC) has updated NUREG-1650, “The United States of America Fifth National Report for the Convention on Nuclear Safety,” Revision 3, issued August 2010, and will submit this report for peer review at the sixth review meeting of the Convention on Nuclear Safety at the International Atomic Energy Agency in Vienna Austria, in March 2014. This report addresses the safety of land-based commercial nuclear power plants in the United States. It demonstrates how the U.S. Government achieves and maintains a high level of nuclear safety worldwide by enhancing national measures and international cooperation, and by meeting the obligations of all the articles established by the Convention. These articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, the priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation. This report addresses the actions the United States has taken to improve nuclear safety in response to the March 11, 2011, accident at the Fukushima Daiichi nuclear power plant in Japan.

Similar to the U.S. National Report issued in 2010, this revised document includes a section developed by the Institute of Nuclear Power Operations describing work that the U.S. nuclear industry has done to ensure safety. The primary responsibility for the safety of a nuclear installation rests with the license holder; therefore, Part 3 explains how the nuclear industry maintains and improves nuclear safety.

CONTENTS

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has prepared Revision 5 to NUREG-1650, "United States of America Sixth National Report for the Convention on Nuclear Safety" for submission for peer review at the sixth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2014. The NRC issued the fifth report in August 2010. This revised report addresses the safety of land-based commercial nuclear power plants in the United States. It demonstrates how the U.S. Government achieves and maintains a high-level of nuclear safety worldwide by enhancing national measures and international cooperation and by meeting the obligations of all the articles established by the Convention. These articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, the priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation.

This report addresses the issues identified through the peer review conducted during the fifth review meeting in April 2010 and discusses challenges and issues that have arisen since that time. The fifth review meeting identified the following U.S. challenges:

(1) addressing buried piping degradation and ground water protection issues
(2) evaluating and responding to cyber security threats
(3) licensing process regarding digital instrumentation and control
(4) enhancing the safety and security interface

The United States highlighted the following planned initiatives at the fifth review meeting:

(1) address Integrated Regulatory Review Service mission recommendations
(2) address buried piping degradation and ground water protection issues
(3) finalize the emergency preparedness rulemaking

This report also discusses the status of safety issues raised in the Fifth U.S. National Report, including reactor materials degradation, cyber security, digital upgrades to instrumentation and control, moisture effects on underground cables, containment pressure credit for emergency core cooling system pump net positive suction head, gas voiding issues in light-water reactor safety systems, and enhancements to emergency preparedness regulations. The report also addresses safety and regulatory issues that have arisen since 2010, such as lessons learned from the March 2011 events at Fukushima.

The Institute of Nuclear Power Operations (INPO) has also provided input to this report. The primary responsibility for the safety of a nuclear installation rests with the license holder; therefore, Part 3 explains how the nuclear industry maintains and improves nuclear safety.

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ABBREVIATIONS

ABWR	advanced boiling-water reactor
ADAMS	Agencywide Documents Access and Management System (NRC)
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	Advanced Passive
ASME	American Society of Mechanical Engineers
BRIIE	Baseline Risk Index for Initiating Events
BWR	boiling-water reactor
BWRVIP	Boiling-Water Reactor Vessel and Internals Project
CEO	chief executive officer
CFR	*Code of Federal Regulations*
CFSI	counterfeit, fraudulent, and suspect items
CNS	Convention on Nuclear Safety
DHS	U.S. Department of Homeland Security
DOE	U.S. Department of Energy
EDG	emergency diesel generator
EGM	enforcement guidance memorandum
EOP	emergency operations procedure
EPA	U.S. Environmental Protection Agency
EPIX	Equipment Performance and Information and Exchange System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Administration
ESBWR	economic simplified boiling-water reactor
FEMA	Federal Emergency Management Agency
FLEX	Diverse and Flexible Coping Strategies
FY	fiscal year
GDC	General Design Criterion
GI	generic issue
GL	generic letter
GNEP	Global Nuclear Energy Partnership
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IN	information notice
INPO	Institute of Nuclear Power Operations
IP	inspection procedure
IRRS	Integrated Regulatory Review Service
ISAP	integrated safety assessment program

ISG	interim staff guidance
ITAAC	inspection(s), test(s), analysis (analyses), and acceptance criterion/criteria
MD	management directive
MWt	megawatt thermal
NANTeL	National Academy for Nuclear Training e-Learning
NCRP	National Council on Radiation Protection and Measurements
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NFPA	National Fire Protection Association
NIMS	National Incident Management System
NRC	U.S. Nuclear Regulatory Commission
NTTF	Near-Term Task Force
OMB	Office of Management and Budget
OSART	Operational Safety Assessment Review Team
POC	performance objectives and criteria
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RASCAL	Radiological Assessment System for Consequence Analysis
RFI	request for information
RG	regulatory guide
RIS	regulatory issue summary
RISC	risk-informed safety class
RS	review standard
SAMGs	severe accident management guidelines
SAT	systems approach to training
SBO	station blackout
SE	safety evaluation
SEE-IN	Significant Event Evaluation and Information Network
SEN	Significant Event Notification
SEP	Systematic Evaluation Program
SFP	spent fuel pool
SOER	significant operating experience report
SSC	structure, system, and component
TI	temporary instruction
TMI	Three Mile Island
TVA	Tennessee Valley Authority
U.S.	United States
US APWR	U.S. Advanced Pressurized Water Reactor
US EPR	U.S. Evolutionary Power Reactor

WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association

PART 1

INTRODUCTION

This section describes the purpose and structure of the “United States of America Sixth National Report for the Convention on Nuclear Safety,” and provides a summary of changes to the sixth United States (U.S.) National Report.

Purpose and Structure of This Report

The United States of America is submitting this updated report for peer review to the sixth review meeting of the Contracting Parties to the Convention on Nuclear Safety (hereafter referred to as the Convention, or CNS). The scope of this report considers only the safety of land-based commercial nuclear power plants, consistent with the definition of nuclear installations provided in Article 2 and the scope of Article 3 of the Convention.

This report demonstrates how the U.S. Government meets the following objectives described in Article 1 of the Convention:

(i) to achieve and maintain a high level of nuclear safety worldwide through the enhancement of national measures and international cooperation including, where appropriate, safety-related technical cooperation

(ii) to establish and maintain effective defenses in nuclear installations against potential radiological hazards to protect individuals, society, and the environment from harmful effects of ionizing radiation from such installations

(iii) to prevent accidents with radiological consequences and to mitigate such consequences should they occur

Technical and regulatory experts from the U.S. Nuclear Regulatory Commission (hereafter referred to as the NRC, Commission,[1] agency, or staff) updated the sixth U.S. National Report, principally using agency information that is publicly available. This updated report follows the format of the Fifth U.S. National Report, and is designed to be a standalone document. Therefore, this report duplicates some of the information presented in the 2010 (fifth) report. To facilitate peer review, Part 1 of this report includes a summary of the main changes to the report (Table 1). This table is followed by a high level summary of the report, consistent with the guidance of the Convention.

Part 2 discusses the Convention’s Articles 6 through 19. Chapters are numbered according to the article of the Convention under consideration. Each chapter begins with the text of the article, followed by an overview of the material covered and a discussion of how the United States meets the obligations described in the article. Articles 6 through 9 summarize the existing nuclear installations and the legislative and regulatory system governing their safety and discuss the adequacy and effectiveness of that system. Articles 10 through 16 address general safety considerations and summarize major safety-related features. Articles 17 through 19 address the safety of installations.

Similar to the 2010 report, Part 3 of this document includes a contribution by the Institute of

1 Commission may also refer to the Chairman and Commissioners who head the NRC.

Nuclear Power Operations (INPO) describing work that the U.S. nuclear industry has done to ensure safety. INPO is a nongovernmental corporation founded in 1979 by the U.S. nuclear industry to collectively promote the highest levels of safety and reliability at U.S. nuclear plants. The primary responsibility for the safety of a nuclear installation rests with the license holder; therefore, Part 3 explains how the nuclear industry maintains and improves nuclear safety.

The report concludes with a series of appendices that discuss the NRC's main challenges as described in the NRC Strategic Plan, the Inspector General's report, the NRC's response to International Atomic Energy Agency's (IAEA) Action Plan on Nuclear Safety, followed by appendices of references, abbreviations, and acknowledgments. The annexes of the report include a list of nuclear plants in the United States, and industry performance indicators.

This report does not explicitly discuss Articles 1 through 5 because the general text of the report, and indeed the very existence of the report, fulfills the requirements of these articles. In accordance with Article 1, the report illustrates how the U.S. Government meets the objectives of the Convention. The report discusses the safety of nuclear installations according to the definition in Article 2 and the scope of Article 3. It addresses implementing measures (such as national laws, legislation, regulations, and administrative means) according to Article 4. Submission of this report fulfills the obligation under Article 5 on reporting. In addition, the information in this report is available in more detail on the NRC's public Web site.

Changes to the Sixth U.S. National Report

To facilitate peer review of this report, Table 1 summarizes the changes to the sixth U.S. National Report. A revision bar along the left margin of the page identifies changes from the fifth report.

Table 1 Summary of Changes to the Sixth U.S. National Report

Report Section	Change
Abstract	Updated to add discussion about the Fifth CNS and Fukushima
Executive Summary	Updated to add discussion about the Fifth CNS and Fukushima
PART 1	
Introduction	Updated to add discussion about the Fifth CNS
Purpose and Structure of This Report	Updated to add discussion about the Fifth CNS and Fukushima
Summary of Changes to the Fifth U.S. National Report	Updated table
Section 1. SUMMARY	New Section. Reordering.
1.1. The U.S. Policy toward Nuclear Activities	Reordering. Editorial changes only
1.1.1 Regulatory Body Organizational Values	Reordering. Updated to add discussion on openness
1.1.2 Regulatory Body Challenges	Reordering. Updated to add discussion on most recent NRC Strategic Plan and Inspector General report
1.2 National Nuclear Programs	Reordering and editorial changes only

1.2.1 Reactor Oversight Process	Reordering. Updated to add discussion about the 2011 self-assessment
1.2.2 License Renewal	Reordering. Updated to add discussion about units entering the 41st year of operation
1.2.3 Power Uprate Program	Reordering. Editorial changes only.
1.2.4 New Reactor Licensing	Updated to add discussion about licenses granted
1.3 Safety and Regulatory Issues, and Regulatory Accomplishments	Reordering. Updated Sections.
1.3.1 Safety and Regulatory Issues Discussed in the Fifth U.S. National Report	Completely updated to add current status
1.3.2 Current Safety and Regulatory Issues	New Section. Discusses 7 new topics, including Fukushima.
1.3.3 Major Regulatory Accomplishments	Updated. Discusses 7 new topics, including Fukushima
1.4 International Peer Reviews and Missions	New section
1.4.1 Convention on Nuclear Safety	Renamed. Updated to include results from Fifth CNS report and Rapporteurs' findings
1.4.2 Integrated Regulatory Review Service	New section. Provide summary results.
1.4.3 Operational Safety Review Team	New section. Provide summary results.
PART 2	
ARTICLE 6. EXISTING NUCLEAR INSTALLATIONS	Editorial changes only
6.1 Introduction	Updated to add safety strategic outcomes in fiscal years 2010-2012
6.2 Nuclear Installations in the United States	Updated to include information on combined licenses issued and decommissioning plants
6.3 Regulatory Processes and Programs	No change
6.3.1 Reactor Licensing	Updated to include information on combined licenses issued
6.3.2 Reactor Oversight Process	Updated to add discussion about the 2011 self-assessment and Action Matrix status
6.3.3 Industry Trends Program	Updated to add discussion about significant events
6.3.4 Accident Sequence Precursor Program	Updated to include a discussion about the accident sequence precursor program status report issued in 2012
6.3.5 Operating Experience Program	Editorial changes only
6.3.6 Generic Issues Program	Updated to add discussion about changes to the program made in 2011
6.3.7 Rulemaking	Updated process
6.3.8 Fire Protection Regulation Program	Updated to add discussion about risk-informed, performance-based fire protection rule and the research program
6.3.9 Decommissioning	Updated to reference relevant regulations and guidance documents

6.3.10 Reactor Safety Research Program	Editorial changes only
6.3.11 Public Participation	Renamed. Updated to refine discussion on petition process
6.4 Lessons Learned from Fukushima	New section
ARTICLE 7. LEGISLATIVE AND REGULATORY FRAMEWORK	Editorial changes only
7.1 Legislative and Regulatory Framework	Editorial changes only
7.2 Provisions of the Legislative and Regulatory Framework	No changes
7.2.1 National Safety Requirements and Regulations	Editorial changes only
7.2.2 Licensing of Nuclear Installations	Updated to refine discussion on 10 CFR Part 52 and hearing process
7.2.3 Inspection and Assessment	Editorial changes only
7.2.4 Enforcement	Editorial changes only
7.3 Lessons Learned from Fukushima	New section
ARTICLE 8. REGULATORY BODY	Editorial changes only
8.1 The Regulatory Body	Editorial changes only
8.1.1 Mandate	No changes
8.1.2 Authority and Responsibilities	No changes
8.1.2.1 Scope of Authority	Editorial changes only
8.1.2.2 The NRC as an Independent Regulatory Agency	Editorial changes only
8.1.3 Structure of the Regulatory Body	Editorial changes only
8.1.3.1 The Commission	Editorial changes only
8.1.3.2 Component Offices of the Commission	Editorial changes only
8.1.3.3 Offices of the Executive Director for Operations	Noted organizational changes. Office of Human Resources renamed.
8.1.3.4 Advisory Committees	Added discussion on Advisory Committee on the Medical Uses of Isotopes
8.1.3.5 Atomic Safety and Licensing Board Panel	Editorial changes only
8.1.3.6 Office of the Inspector General	Reordering. Updated description.
8.1.4 Position of the NRC in the Governmental Structure	Renumbered. No changes.
8.1.4.1 Executive Branch	Renumbered. Editorial changes only.
8.1.4.2 The States (i.e., of the United States)	Renumbered. Editorial changes only.
8.1.4.3 Congress	Renumbered. Editorial changes only.

8.1.5 International Responsibilities and Activities	Renumbered. Updated to discuss missions and Fukushima
8.1.5.1 International Standards	New section
8.1.5.2 Integrated Regulatory Review Service Mission	Renumbered. Updated to add discussion on findings and followup mission
8.1.5.3 Operational Safety Assessment Review Teams	New section
8.1.6 Financial and Human Resources	Renumbered.
8.1.6.1 Financial Resources	Renumbered. Updated to add funds for fiscal years 2010-2011
8.1.6.2 Human Resources	Renumbered. Updated to discuss survey findings
8.1.7 Openness and Transparency	New section
8.2 Separation of Functions of the Regulatory Body from Those of Bodies Promoting Nuclear Energy	Editorial changes only
8.3 Fukushima Lessons Learned	New section
ARTICLE 9. RESPONSIBILITY OF THE LICENSE HOLDER	Editorial changes only
9.1 Introduction	Editorial changes only
9.2 The Licensee's Primary Responsibility for Safety	No changes
9.3 NRC Enforcement Program	Updated to reference new policy and discuss recent enforcement actions
9.4 Openness and Transparency	New section
9.5 Fukushima Lessons Learned	New section
ARTICLE 10. PRIORITY TO SAFETY	Editorial changes only
10.1 Background	Editorial changes only
10.2 Probabilistic Risk Assessment Policy	Editorial changes only
10.3 Applications of Probabilistic Risk Assessment	Updated to provide new references
10.3.1 Risk-Informed Special Treatment	Updated to discuss pilot application
10.3.2 Risk-Informed Inservice Inspection	Updated to provide new references
10.3.3 Risk-Informed Technical Specification Changes	Updated initiatives
10.3.4 Development of Standards	Editorial changes only
10.4 Safety Culture	Updated to discuss new policy.
10.4.1 NRC Monitoring of Licensee Safety Culture	No changes

10.4.1.1 Background	No changes
10.4.1.2 Enhanced Reactor Oversight Process	Updated to discuss common safety culture language.
10.4.2 The NRC Safety Culture	Updated surveys' information
10.5 Managing the Safety and Security Interface	Updated to add discussion on integration of cornerstone
10.6 Fukushima Lessons Learned	New section
ARTICLE 11. FINANCIAL AND HUMAN RESOURCES	Editorial changes only
11.1 Financial Resources	Editorial changes only
11.1.1 Financial Qualifications Program for Construction and Operations	No changes
11.1.1.1 Construction Permit Reviews	No changes
11.1.1.2 Operating License Reviews	No changes
11.1.1.3 Combined License Application Reviews	No changes
11.1.1.4 Postoperating License Nontransfer Reviews	No changes
11.1.1.5 Reviews of License Transfers	No changes
11.1.2 Financial Qualifications Program for Decommissioning	Editorial changes only
11.1.3 Financial Protection Program for Liability Claims Arising from Accidents	Editorial changes only
11.1.4 Insurance Program for Onsite Property Damages Arising from Accidents	No changes
11.2 Regulatory Requirements for Qualifying, Training, and Retraining Personnel	No changes
11.2.1 Governing Documents and Process	Updated to add discussion on requalification
11.2.2 Experience	Updated
11.3 Fukushima Lessons Learned	New section
ARTICLE 12. HUMAN FACTORS	Editorial changes only
12.1 Goals and Mission of the Program	No changes
12.2 Program Elements	Shortened
12.3 Significant Regulatory Activities	No changes
12.3.1 Human Factors Engineering Issues	New references and experience provided
12.3.2 Emergency Operating Procedures and Plant Procedures	New references and experience provided
12.3.3 Shift Staffing	New references provided and discussion on new reactors

12.3.4 Fitness for Duty	Updated to add discussion about the fatigue management rulemaking
12.3.5 Human Factors Information System	Editorial changes only
12.3.6 Support to Event Investigations and For-Cause Inspections and Training	Updated to add discussion on recent inspections
12.4 Fukushima Lessons Learned	New section
ARTICLE 13. QUALITY ASSURANCE	Editorial changes only
13.1 Background	No changes
13.2 Regulatory Policy and Requirements	Editorial changes only
13.2.1 Appendix A to 10 CFR Part 50	Editorial changes only
13.2.2 Appendix B to 10 CFR Part 50	Editorial changes only
13.2.3 Approaches for Adopting More Widely Accepted International Quality Standards	No changes
13.3 Quality Assurance Regulatory Guidance	No changes
13.3.1 Guidance for Staff Reviews for Licensing	No changes
13.3.2 Guidance for Design and Construction Activities	New references added
13.3.3 Guidance for Operational Activities	New references added
13.4 Quality Assurance Programs	New references added
13.5 Quality Assurance Audits Performed by Licensees	No changes
13.5.1 Audits of Vendors and Suppliers	No changes
13.6 Fukushima Lessons Learned	New section
ARTICLE 14. ASSESSMENT AND VERIFICATION OF SAFETY	Editorial changes only
14.1 Ensuring Safety Assessments throughout Plant Life	No changes
14.1.1 Assessment of Safety	New section.
14.1.2 Maintaining the Licensing Basis	Reordered. Editorial changes only.
14.1.2.1 Governing Documents and Process	Reordered. Editorial changes only.
14.1.1.2 Regulatory Framework for the Restart of Browns Ferry Unit 1	Deleted. Section no longer provides relevant information.
14.1.3 Power Uprates	New section.
14.1.3.1 Governing Documents and Process	New section.
14.1.3.2 Experience	New section.
14.1.4 License Renewal	Reordered.
14.1.4.1 Governing Documents and Process	Reordered. Updated to add discussion on waste confidence and environmental rule.

14.1.4.2 Experience	Reordered. Updated to add discussion about renewed license to date.
14.1.4.3 Operating Beyond 60 Years	Reordered. Updated info on workshops and research
14.1.5 The United States and Periodic Safety Reviews	Reordered. Editorial changes only.
14.1.5.1 The NRC's Robust and Ongoing Regulatory Process and the Current Licensing Basis	Reordered. Updated information on inspection hours.
14.1.5.2 The Backfitting Process: Timely Imposition of New Requirements	Reordered. Updated information on types of backfit.
14.1.5.3 The NRC's Extensive Experience with Broad-Based Evaluations	Reordered. No changes.
14.1.5.4 License Renewal Confirms Safety of Plants	Reordered. No changes.
14.1.5.5 Risk-Informed Regulation and the Reactor Oversight Process	Reordered. Updated information on NFPA.
14.1.5.6 Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations	Reordered. No changes.
14.1.5.7 The NRC's Regulatory Process Compared with International Safety Reviews	Reordered. Editorial changes only.
14.2 Verification by Analysis, Surveillance, Testing, and Inspection	Updated to remove discussion about performance measure and aging management.
14.3 Fukushima Lessons Learned	New section.
ARTICLE 15. RADIATION PROTECTION	Editorial changes only.
15.1 Authorities and Principles	New references provided
15.2 Regulatory Framework	Editorial changes only
15.3 Regulations	Updated to add discussion about interaction with stakeholders and the evaluation of international standards
15.4 Radiation Protection Activities	No changes
15.4.1 Control of Radiation Exposure of Occupational Workers	Updated collective doses
15.4.2 Control of Radiation Exposure of Members of the Public	Updated information on ground water contamination
15.5 Fukushima Lessons Learned	New section
ARTICLE 16. EMERGENCY PREPAREDNESS	Editorial changes only
16.1 Background	New reference provided

16.2 Offsite Emergency Planning and Preparedness	Updated to add discussion on emergency planning steering committee
16.3 Emergency Classification System and Emergency Action Levels	Updated to discuss emergency operating procedures and severe accident management guidelines
16.4 Recommendations for Protective Action in Severe Accidents	Updated completely
16.5 Inspection Practices - Reactor Oversight Process for Emergency Preparedness	Expanded discussion on exercise evaluation
16.6 Responding to an Emergency	No changes
16.6.1 Federal Response	No changes
16.6.2 Licensee, State, and Local Response	No changes
16.6.3 The NRC's Response	Editorial changes only
16.6.4 Aspects of Security that Support Response	Updated to discuss security orders
16.7 Communications with Neighboring States and International Arrangements	Renamed. Expanded discussion. Provided new dates for agreements.
16.8 Communications with the Public	New section
16.9 Fukushima Lessons Learned	New section
ARTICLE 17. SITING	Editorial changes only
17.1 Background	Shortened
17.2 Safety Elements of Siting	Editorial changes only
17.2.1 Background	Editorial changes only
17.2.2 Assessments of Nonseismic Aspects of Siting	New section
17.2.3 Assessments of Seismic and Geological Aspects of Siting	Renumbered. Updated to expand discussion about seismic designs in new reactors and provide new references
17.2.4 Assessments of Radiological Consequences from Postulated Accidents	Renumbered. New references provided.
17.3 Environmental Protection Elements of Siting	Editorial changes only
17.3.1 Governing Documents and Process	Editorial changes only
17.3.2 Other Considerations for Siting Reviews	Added discussion on environmental impact statement
17.4 Reevaluation of Site-Related Factors	New section
17.5 Consultation with other Contracting Parties To Be Affected by the Installation	Renumbered. No changes
17.6 Fukushima Lessons Learned	New section

ARTICLE 18. DESIGN AND CONSTRUCTION	Editorial changes only
18.1 Defense-in-Depth Philosophy	No changes
18.1.1 Governing Documents and Process	Editorial changes only
18.1.2 Experience	Editorial changes only. Added discussion on Watts Bar
18.1.2.1 Regulatory Framework for the Reactivation of Watts Bar Unit 2	Section removed. Merged with 18.1.2. Current status provided.
18.1.2.2 Design Certifications	Deleted section. Not relevant to subject. Repeated information.
18.2 Technologies Proven by Experience or Qualified by Testing or Analysis	No changes
18.3 Design for Reliable, Stable, and Easily Manageable Operation	No changes
18.3.1 Governing Documents and Process	Updated references
18.3.2 Experience	No changes
18.3.2.1 Human Factors Engineering	Program weaknesses addressed.
18.3.2.2 Digital Instrumentation and Controls	Completely updated
18.3.2.3 Cyber Security	Memorandum of understanding discussed
18.4 New Reactor Construction Experience Program	Lessons learned discussed
18.5 Fukushima Lessons Learned	New section
ARTICLE 19. OPERATION	Editorial changes only
19.1 Initial Authorization to Operate	ASLB discussion updated
19.2 Definition and Revision of Operational Limits and Conditions	New references provided
19.3 Approved Procedures	Editorial changes only
19.4 Procedures for Responding to Anticipated Operational Occurrences and Accidents	No changes
19.5 Availability of Engineering and Technical Support	No changes
19.6 Incident Reporting	New references provided
19.7 Programs To Collect and Analyze Operating Experience	Updated to add discussion about INPO’s network
19.8 Radioactive Waste	Updated to add discussion about Blue Ribbon Commission
19.9 Fukushima Lessons Learned	New section
PART 3	

Convention on Nuclear Safety Report: The Role of the Institute of Nuclear Power Operations in Supporting the U.S. Commercial Nuclear Power Industry's Focus on Nuclear Safety	Updated
APPENDIX A NRC STRATEGIC PLAN 2008-2013	Updated to add most recent and updated Strategic Plan
APPENDIX B NRC MAJOR MANAGEMENT CHALLENGES FOR THE FUTURE	Updated to add most recent report from the Inspector General
APPENDIX C U.S. SUPPORT OF THE INTERNATIONAL ATOMIC ENERGY AGENCY ACTION PLAN ON NUCLEAR SAFETY	New Section
APPENDIX D REFERENCES	Renumbered. Updated
APPENDIX E ABBREVIATIONS	Renumbered. Updated
APPENDIX F ACKNOWLEDGMENTS	Renumbered. Updated
ANNEX 1 U.S. COMMERCIAL NUCLEAR POWER REACTORS	Updated
ANNEX 2 U.S. NUCLEAR ELECTRIC INDUSTRY PERFORMANCE INDICATOR GRAPHS	Updated

SECTION 1. SUMMARY

The Summary in the National Report should highlight the Contracting Party's continued efforts in achieving the Convention's objectives. It should serve as a major information source by summarizing updated information on matters that have developed since the previous National Report, focusing discussion on significant changes in national laws, regulations, administrative arrangements, and practices related to nuclear safety, and demonstrating followup from one Review Meeting to the next.

This section provides a high level summary of U.S. policy toward safety; the regulatory body's organizational values, including transparency; and its challenges. It summarizes the national nuclear programs; provides an update on important safety and regulatory issues identified in the previous National Report; and addresses those safety and regulatory issues and regulatory accomplishments that have arisen since the last National Report was issued (see NUREG-1650, "The United States of America Fifth National Report for the Convention on Nuclear Safety," Revision 3, issued August 2010). Lastly, this section summarizes the results of international peer reviews and missions.

1.1 <u>The U.S. Policy Toward Nuclear Activities</u>

The Energy Reorganization Act of 1974 created the U.S. Nuclear Regulatory Commission (NRC) as an independent agency of the Federal Government. The agency's mission is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment. In addition, the agency's export licensing and domestic safeguards programs are integral to the U.S. Government's commitment to nuclear nonproliferation. The NRC's safety and security responsibilities stem from the Atomic Energy Act of 1954, as amended. The agency accomplishes its mission by licensing and overseeing nuclear reactor operations and other activities that apply to the possession of nuclear materials and wastes, ensuring that nuclear materials and facilities are safeguarded from theft and radiological sabotage, issuing rules and standards, inspecting nuclear facilities, and enforcing regulations.

1.1.1 Regulatory Body Organizational Values

In conducting its work, the NRC adheres to seven organizational values to guide its actions: integrity, service, openness, commitment, cooperation, excellence, and respect. The principles of good regulation help carry out NRC regulatory activities. These principles focus on ensuring safety and security while appropriately balancing the interests of stakeholders, including the public and licensees. These principles are independence, efficiency, clarity, reliability, and openness. The NRC's final decisions are based on objective, technical assessments of all information, and are documented with reasons explicitly stated. As a learning organization, the NRC establishes ways to evaluate and continually upgrade its regulatory capabilities. Its regulations are coherent, logical, practical, and based on the best available knowledge from research and operational experience.

The NRC also views nuclear regulation as a service to the public and, as such, it must be transacted openly. The NRC is committed to being a trusted, independent, transparent, and effective regulator. The NRC issuance of its Open Government Plan, first published April 7, 2010, is a reflection of the agency's long history of, and commitment to, openness with the public and transparency in the regulatory process. The agency's goal to ensure openness explicitly recognizes that the public must be informed about, and have a reasonable opportunity to participate meaningfully in, the regulatory process. Except for proprietary information, security-related information, sensitive predecisional information, and sensitive information supplied by foreign governments that is deemed to be sensitive, the NRC makes the documentation that it uses in its decisionmaking process available in the agency's Public Document Room in Rockville, MD, and on the agency's public Web site at http://www.nrc.gov. Over the past several years, the NRC has also embraced social media as an important new tool for reaching a wider public audience. As a result, a significant amount of information about nuclear activities and the national policy regarding them is available to everyone.

1.1.2 Regulatory Body Challenges

The NRC identified major challenges for the future in its Strategic Plan for 2008-2013, updated in February 2012. External factors may cause changes to the regulatory environment. To adapt to these changes, the NRC must use its resources efficiently, revise the regulatory framework as appropriate to disposition existing or emerging issues, and provide adequate infrastructure to maintain staff competence and readiness. Some expected changes include:

- The NRC expects to continue to receive additional applications from entities that want to build and operate small and large new nuclear power plants.

- Several additional sites will be entering a decommissioning phase – requiring a modification in licensing and inspection.

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- The NRC will enhance, where necessary, its regulatory system and incorporate lessons learned from the March 2011, accident at the Fukushima facility in Japan while ensuring the most important operational issues remain the agency's top priority.

- The NRC will increase its international engagement including enhancing the lessons learned from Fukushima and continue to engage emerging nations just entering the commercial nuclear regime.

The NRC will continue to coordinate with Federal, State, local, and Tribal authorities on a wide variety of issues related to emergency planning and the safety and security of nuclear facilities and materials while maintaining effective and open communication with public stakeholders on these and other issues.

As stated in the Strategic Plan, the following key external factors could affect the agency's ability to achieve its strategic goals:

- a significant operating incident (domestic or international)
- a significant terrorist incident
- emergency preparedness and incident response
- legislative initiatives
- international nuclear safety developments
- international treaties and conventions

- major revision to the law on nuclear waste

By law, the Inspector General of each Federal agency (as discussed in Article 8) identifies the agency's most serious management and performance challenges and assesses progress in addressing them. The NRC's Inspector General's annual assessment of the major management challenges confronting the agency appear on the NRC's public Web site. The 2012 assessment report described the following main challenges, given in more detail in Appendix B to this report.

- management of regulatory processes to meet a changing environment in the oversight of nuclear materials
- management of internal NRC security and oversight of licensee security programs
- management of regulatory processes to meet a changing environment in the oversight of nuclear facilities
- management of issues associated with the safe storage of high-level radioactive waste when there is no long-term disposal solution
- management of information technology
- administration of all aspects of financial management and procurement
- managing human capital

1.2 National Nuclear Programs

The NRC has a number of programs and processes to protect public health and safety and the environment and to meet the obligations of the Convention on Nuclear Safety (CNS). Key programs and processes in the reactor arena comprise a well-established licensing process, which includes: (1) reactor oversight, (2) license renewal, (3) power uprates, and (4) new reactor licensing. As described in Section 10.5 of this report, the security cornerstone was reintegrated into the Reactor Oversight Process in 2012.

1.2.1 Reactor Oversight Process

The NRC's Reactor Oversight Process is now 12 years old. The Reactor Oversight Process focuses on cornerstones of safety, such as initiating events, public radiation safety, emergency preparedness, and security, which are assessed through a combination of performance indicators and risk-informed inspections.

In its annual self-assessment for calendar year 2012, the NRC staff concluded that the Reactor Oversight Process provided effective safety oversight as demonstrated by meeting the program goals and achieving its intended outcomes. The self-assessment demonstrated that the Reactor Oversight Process was successful in being objective, risk-informed, understandable, and predictable. It also showed that the Reactor Oversight Process ensures openness and effectiveness in support of the agency's mission and its strategic goals of safety and security. The NRC appropriately monitored operating nuclear power plant activities and focused agency resources on performance issues and plants continued to receive a level of oversight commensurate with their performance. The Reactor Oversight Process has developed into a mature oversight process over the past 12 years. However, the NRC recognizes the value of continuous improvement; therefore, it actively solicits stakeholder feedback to apply lessons learned and improve various aspects of the Reactor Oversight Process.

Inspection reports, including the results of emergency exercise evaluations, are on the NRC public Web site at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/listofrpts_body.html. Article 6 of this report discusses the Reactor Oversight Process in detail.

1.2.2 License Renewal

The NRC's review of license renewal applications focuses on maintaining plant safety and particularly considers the effects of aging on important structures, systems, and components. The review of a renewal application proceeds along two paths -- one to review safety issues and the other to assess potential environmental impacts. Applicants must demonstrate that they have identified and can manage the effects of aging and can continue to maintain an acceptable level of safety throughout the period of extended operation. Applicants must also address the environmental impacts from extended operation. The Commission has seen sustained, strong interest in license renewal, which allows plants to operate up to 20 years beyond their current operating licenses. The Atomic Energy Act established the original 40-year term, a timeframe based on economic and antitrust considerations, rather than the technical limitations of the nuclear facility.

The decision to seek license renewal is voluntary and rests entirely with nuclear power plant owners. The decision typically is based on the plant's economic viability and whether it can continue to meet the Commission's requirements. Currently, approximately three quarters of the plants in the United States have had their operating licenses renewed. Based on statements from industry representatives, the Commission expects nearly all sites to apply for license renewal. As reported in the Fifth U.S. National Report, seven units entered their 41st year of operation (the period of extended operation) in 2009 and 2010. By the end of 2013, 18 additional units will have entered the period of extended operation as listed below.

Year 2011	Year 2012	Year 2013[2]
• Dresden Unit 3 • Palisades Unit 1	• Pilgrim Unit 1 • Quad Cities Unit 1 • Quad Cities Unit 2 • Surry Unit 1 • Turkey Point Unit 3 • Vermont Yankee Unit 1	• Browns Ferry Unit 1 • Fort Calhoun Unit 1 • Indian Point Unit 2 • Oconee Unit 1 • Oconee Unit 2 • Peach Bottom Unit 2 • Point Beach Unit 2 • Prairie Island Unit 1 • Surry Unit 2 • Turkey Point Unit 4

See Section 6.2 of this report for additional discussion on the Kewaunee Power Station, Crystal River Unit 3, San Onofre Nuclear Generating Station Units 2 and 3, and Vermont Yankee Nuclear Power Station licenses status. Article 14 of this report discusses the license renewal process in detail, including a discussion of the upcoming update to the Generic Environmental Impact Statement for license renewal.

2 On October 22, 2012, Dominion Resources, the operator of Kewanee Power Station, announced that it would close the plant and move to safe shutdown in the second quarter of 2013, lowering the number of units entering the period of extended operation in 2013 from 11 to 10. The station will be under NRC oversight throughout the decommissioning process.

1.2.3 Power Uprates

Under its licensing program, the NRC carefully reviews requests to raise the maximum thermal power level at which a plant may be operated. In reviewing these power uprate requests, NRC's review focuses on safety. The agency closely monitors operating experience to identify safety issues that may affect the implementation of power uprates.

Power uprates can be classified as: (1) measurement uncertainty recapture power uprates, (2) stretch power uprates, and (3) extended power uprates. Measurement uncertainty recapture power uprates are less than a 2 percent increase in power and are achieved by implementing higher precision feedwater flow measurement devices to more accurately calculate reactor power. Stretch power uprates have increased power up to 7 percent and are generally within the original design capacity of the plant. Stretch power uprates usually involve changes to instrumentation setpoints and generally do not involve major plant modifications. Extended power uprates usually increase power more than 7 percent and require significant modifications to major balance-of-plant equipment. The NRC has approved extended power uprates of up to 20 percent.

Article 14 of this report discusses the power uprate process in detail.

1.2.4 New Reactor Licensing

The NRC New Reactor Program consists of three subprograms: licensing, construction inspection, and advanced reactors. The NRC is focusing on completing ongoing licensing reviews, supporting construction activities associated with five new reactor units in the United States (one unit licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and undergoing construction completion, and four new units licensed under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants") and positioning itself for success in the advanced reactor program by investing in activities to establish the necessary regulatory framework and infrastructure for advanced reactors. The NRC's new reactor program is also actively engaged in several international cooperative activities to promote enhanced safety in new reactor designs, strengthen reactor siting reviews, and improve the effectiveness and efficiency of inspections and the collection and sharing of construction experience.

The NRC staff is interacting with vendors and utilities on new reactor applications and licensing activities. The NRC staff is actively reviewing 10 combined license applications and 1 early site permit application, in addition to 3 design certification applications. Five combined license applicants have requested that the NRC suspend its review of their applications given changing business strategies. All combined license applicants are using the licensing process specified in 10 CFR Part 52, which is designed to be more stable and predictable than the process specified in 10 CFR Part 50. This licensing process resolves all safety and environmental issues, as well as emergency preparedness and security issues, before a new nuclear power plant is constructed.

To date, the NRC staff has issued design certifications for four reactor designs that can be referenced in an application for a nuclear power plant: (1) General Electric Nuclear Energy's Advanced Boiling Water Reactor (ABWR), (2) Westinghouse Electric Company, LLC's (Westinghouse's) System 80+, (3) Westinghouse's Advanced Passive (AP) 600 design, and (4)

Westinghouse's AP1000. In December 2011, the NRC staff issued amendments to the AP1000 and ABWR design certification rules

The NRC staff is currently performing the following design certification reviews: (1) General Electric Hitachi Nuclear Energy's Economic Simplified Boiling-Water Reactor (ESBWR), (2) AREVA Nuclear Power's U.S. Evolutionary Power Reactor (US EPR), and (3) Mitsubishi Heavy Industries, Ltd.'s U.S. Advanced Pressurized-Water Reactor (US APWR). In addition, the NRC staff has received two applications to renew the ABWR design certification from General Electric Hitachi Nuclear Energy and from Toshiba Corporation. In late-2013, the NRC is expecting to receive a design certification application for the APR-1400 reactor design sponsored by Korea Hydro and Nuclear Power Company.

By certifying nuclear reactor designs, the NRC resolves safety issues in a design certification rulemaking. When an applicant submits an application for construction of a new nuclear power plant using one of the certified designs, the license application review can proceed more efficiently in a way that ensures safety while minimizing unnecessary regulatory burden and delays.

In 2012, the NRC issued its first combined licenses authorizing construction and operation of new nuclear power plants at two sites in the U.S. The NRC issued combined licenses referencing the AP1000 certified design to Southern Nuclear Operating Company for two units at Vogtle in Georgia and to South Carolina Electric and Gas Company and South Carolina Public Service Authority for two units at V.C. Summer in South Carolina.

The NRC staff is actively reviewing nine combined license applications for a total of 15 new nuclear power plants at the following sites: (1) Levy County in Florida, (2) William States Lee III in South Carolina, (3) Turkey Point in Florida, (4) Fermi Unit 3 in Michigan, (5) South Texas Project in Texas, (6) Calvert Cliffs in Maryland, (7) Bell Bend in Pennsylvania, (8) Comanche Peak in Texas, and (9) North Anna in Virginia. The NRC has suspended review of the combined license applications, at the applicants' requests, for the following sites: (1) Bellefonte Units 3 and 4 in Alabama, (2) Callaway in Missouri, (3) Grand Gulf in Mississippi, (4) Nine Mile Point in New York, (5) River Bend Station in Louisiana, and (6) Shearon Harris in North Carolina. In April 2013, Dominion Virginia Power switched its nuclear technology selection for the North Anna Unit 3 COL project from the US APWR to the ESBWR design.

To date, the NRC has issued four early site permits to the following applicants: (1) System Energy Resources, Inc., for the Grand Gulf site in Mississippi; (2) Exelon Generation Company, LLC, for the Clinton site in Illinois; (3) Dominion Nuclear North Anna, LLC, for the North Anna site in Virginia; and (4) Southern Nuclear Operating Company for the Vogtle Electric Generating Plant in Georgia. These are the first early site permits that the NRC has issued and the first time this portion of the 10 CFR Part 52 licensing process was implemented. According to this process, environmental issues that have been resolved in the early site permit proceedings cannot be re-opened during a combined license proceeding.

The NRC staff is currently reviewing one early site permit application submitted by the Public Service Enterprise Group on May 25, 2010, for a site in Salem County, New Jersey. The application uses a plant parameter envelope methodology because a reactor technology has not yet been selected. On August 28, 2012, Exelon Nuclear Texas Holding, LLC, formally withdrew an early site permit application for the Victoria County Station Site in Texas, citing changes in its business plans.

Following the issuance of the combined licenses to Southern Nuclear Operating Company on February 10, 2012, for two AP1000 units at the Vogtle site, and to South Carolina Electric and Gas Company on March 30, 2012, for two AP1000 units at the V.C. Summer site, the pace of construction inspection significantly increased. The NRC increased the staff at both Vogtle and the V.C. Summer construction resident inspector offices. The NRC has a dedicated construction inspection organization in its Region II office in Atlanta, Georgia, that carries out all construction inspection activities across the United States, including the day-to-day onsite inspections and the specialized inspections needed to support NRC oversight of the construction of new nuclear power plants.

One partially built plant, Watts Bar Nuclear Plant Unit 2, had stopped construction activities in the mid-1980s. Watts Bar Unit 2 is a Westinghouse-designed pressurized-water reactor (PWR) located in Tennessee and owned by the Tennessee Valley Authority (TVA). TVA has resumed construction activities and is currently pursuing an operating license approval under 10 CFR Part 50.

In addition to working on domestic issues for new reactor construction, the NRC has been a leader in cooperating with other national nuclear regulatory authorities to address reactor licensing activities. The NRC is a founding member of, and fully participates in, the Multinational Design Evaluation Program, a unique international forum with members from the regulatory authorities of Canada, China, Finland, France, India, Japan, the Republic of Korea, the Russian Federation, South Africa, Sweden, the United Kingdom, and the United States. The regulator from the United Arab Emirates is an associate member of the Multinational Design Evaluation Program, taking part in the APR1400 design specific working group and issue-specific groups, and the International Atomic Energy Agency (IAEA) is involved in the generic work of the program. The Nuclear Energy Agency (NEA) from the Organisation for Economic Co-operation and Development performs the technical secretariat duties for the Multinational Design Evaluation Program.

The activities of the Multinational Design Evaluation Program include: (1) cooperation on specific safety design reviews of Westinghouse Electric Company's AP1000, Korea Hydro and Nuclear Power's APR1400, and AREVA's EPR, and (2) exploration of opportunities to harmonize and converge on regulatory practices in the areas of safety goals, safety classification, digital instrumentation and controls, mechanical codes and standards, and vendor inspection cooperation.

The Multinational Design Evaluation Program interacts with various representatives from the industry, including vendors and operators, standards development organizations, and the World Nuclear Association. The Program also interacts with other regulatory-related bodies such as the NEA's Committee for Nuclear Regulatory Activities and Committee for Safety of Nuclear Installations and the Western European Nuclear Regulators Association.

Articles 17 and 18 of this report discuss new reactor licensing in more detail.

1.3 Safety and Regulatory Issues, and Regulatory Accomplishments

This section provides an update on important safety and regulatory issues identified in the Fifth U.S. National Report, and addresses those safety and regulatory issues and regulatory accomplishments that have arisen since the last National Report was issued

1.3.1 Safety and Regulatory Issues Discussed in the Fifth U.S. National Report

In the Fifth U.S. National Report, the NRC staff and the licensees reported to be working with the safety and regulatory issues listed in this section. An update on the following items is provided:

- reactor materials degradation
- cyber security
- digital upgrades to instrumentation and control
- moisture effects on underground cables
- containment pressure credit for emergency core cooling system pump net positive suction head
- gas voiding issues in light-water reactor safety systems
- enhancements to emergency preparedness regulations

Reactor Materials Degradation Issues

Cases involving materials degradation include the degradation of buried piping systems and the degradation of neutron-absorber materials in spent fuel pools.

Degradation of Buried Piping Systems

Over the past several years, instances of buried piping leaks have occurred in safety-related and nonsafety-related piping at nuclear power plants. Most of the leaks have occurred in nonsafety-related piping. Some of these leaks have caused inadvertent releases of low-level radioactive material and diesel fuel oil to the environment. The pipe degradation leading to these leaks has not affected the operability of safety systems. The type and amount of radioactive material released to the environment has been a small fraction of the regulatory limits. Consequently, these pipe leaks are of low significance with respect to public health and safety. These events have, however, resulted in significant public interest. As a result of these leaks, the NRC has internally evaluated the technical and regulatory significance of these events and the nuclear industry has developed a voluntary initiative intended to "achieve assurance of structural and leakage integrity of in-scope components."

The NRC's internal review of issues associated with leaks from buried piping is substantially documented in communications between the NRC staff and the Commission (SECY-09-0174, "Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities," dated December 2, 2009; SRM SECY-11-0019, "Senior Management Review of Overall Regulatory Approach to Groundwater Protection" and SRM SECY-11-0076, "Improving the Public Radiation Safety Cornerstone of the Reactor Oversight Process"). These reviews resulted in the creation of the NRC's "Buried Piping Action Plan," updated in November 2011, which determined that (1) existing regulations were sufficient to provide reasonable assurance that the structural integrity of buried piping would be maintained and that leakage of radioactive material from buried piping would remain within regulatory limits; (2) the radiological effluent performance indicator currently in the Reactor Oversight Program should not be revised; and (3) the industry initiative on buried piping should not be incorporated into regulation; instead, industry activities should be monitored to ensure that the initiative is being implemented in a "committed and enduring fashion." To implement these determinations, the staff participates in industry meetings such as the Electric Power Research Institute's (EPRI) Buried Piping Integrity Group;

participates in the development of industry standards (i.e., American Society of Mechanical Engineers (ASME) and NACE International) related to buried piping at nuclear power plants; and developed and is implementing a Temporary Instruction (TI) 2515/182, “Review of Implementation of the Industry Initiative to Control Degradation of Underground Piping and Tanks,” issued November 2011, to monitor the implementation of the industry buried pipe initiative.

The U.S. nuclear industry developed the Buried Piping and the Underground Piping and Tanks Integrity Initiatives. The staff periodically meets with the industry to further understand these initiatives, evaluate their effectiveness, and monitor industry implementation. The NRC remains closely engaged with the nuclear industry on the implementation and modification of these initiatives. The Commission directed the staff to oversee the initiative to ensure implementation in a “committed and enduring fashion.” NRC inspections of plant underground piping initiative programs have shown that plants are implementing the initiatives in a “committed and enduring fashion” as directed by the Commission.

Based on the NRC’s internal review of buried piping issues, and the TI inspection results, no changes to the regulatory framework regarding buried pipe are currently being contemplated.

Degradation of Neutron-Absorber Materials in Spent Fuel Pools

One of the NRC’s strategic outcomes for its safety goal is that there are “no inadvertent criticality events.” To achieve this goal, as it relates to the storage and handling of reactor fuel, the NRC has issued regulations focused on maintaining spent fuel pools (SFPs) subcritical under normal and accident conditions. These regulations appear in 10 CFR 50.68, “Criticality Accident Requirements,” and General Design Criterion 62, “Prevention of Criticality in Fuel Storage and Handling,” in Appendix A to 10 CFR Part 50. To satisfy these regulations, most licensees have installed fixed neutron-absorbers and neutron-absorbing inserts in the SFP storage racks. Degradation or deformation of the credited neutron absorbing materials could reduce the material’s ability to perform its safety function and potentially violate the NRC’s subcriticality regulations.

There are many different types of neutron absorbing materials. The most common types of neutron absorbing materials in United States SFPs are Boraflex, Carborundum, Boral, and Metamic. Boraflex was the first neutron-absorbing material to exhibit significant degradation. The NRC documented this issue in Information Notice (IN) 87-43, “Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks,” dated September 8, 1987; IN 93-70, “Degradation of Boraflex Neutron Absorber Coupons,” dated September 10, 1993; and IN 95-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Pool Storage Racks,” dated September 8, 1995; and in Generic Letter (GL) 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks,” dated June 26, 1996. Ultimately, this issue was resolved through either revised plant-specific criticality analyses that reduced or eliminated credit for Boraflex or by the replacement of Boraflex with other neutron-absorbing materials.

Although the Boraflex degradation issue was previously addressed, issues with monitoring and mitigating of the degradation of Boraflex are still being identified as documented in IN 2012-13, “Boraflex Degradation Surveillance Programs and Corrective Action in the Spent Fuel Pool,” dated August 10, 2012. Subsequent to this operating experience, two Technical Letter Reports were released pertaining to some of the methods used to monitor the degradation of neutron absorbing materials: “Boraflex, RACKLIFE, and BADGER, Description and Uncertainties” and

“Initial Assessment of Uncertainties Associated with BADGER Methodology,” both issued in September 2012. These reports identify gaps in knowledge and uncertainties in these surveillance methodologies. Additional reports pertaining to other aspects of neutron absorbing-material degradation will be issued in the near future.

Recent operating experience has identified several instances of degradation, deformation, or both of Carborundum and Boral neutron-absorbing materials in the SFPs of operating reactors. One example of neutron-absorbing material degradation occurred in the Palisades Nuclear Plant. On July 15, 2008, in support of its license renewal activities, the licensee performed “blackness testing” of the SFP racks to verify its Carborundum was performing in accordance with the assumptions in its criticality analysis of record. Based on this testing, the licensee could not confirm that the SFP met the subcriticality requirements in 10 CFR 50.68 or its technical specifications. Since the licensee did not have an established monitoring program for the Carborundum, the onset of the degradation and the degradation rate cannot be established. The safety significance of this finding is that the licensee’s SFP was in an unanalyzed condition and, although margin remained, the amount of margin to the pool having a criticality event was not known. In response, the NRC issued IN 2009-26, “Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool,” dated October 23, 2009, and License Renewal Interim Staff Guidance (ISG) 2009-01, “Staff Guidance Regarding Plant-Specific Aging Management Review and Aging Management Program for Neutron-Absorbing Material in Spent Fuel Pools,” dated November 23, 2009, and NUREG 1801, “Generic Aging Lessons Learned (GALL) Report,” Revision 2, issued December 2010.

The NRC has begun to evaluate the regulatory changes that may be necessary to ensure that its licensees can identify and mitigate neutron-absorber degradation before it challenges subcriticality safety margins. The Palisades operating experience has highlighted the importance of an effective surveillance program for the early identification of neutron-absorber degradation. Such a program could consist of various testing and identification methods, including, but not necessarily limited to, coupon sampling, in-situ testing, and validated and verified predictive analytical computer codes.

Cyber Security

Following the terrorist attacks on September 11, 2001, the NRC issued a series of advisories, orders, and rulemakings requiring nuclear power plants to enhance cyber security. Although the terrorist attacks did not have a cyber component, an increasing number of nuclear power plants are using digital control systems that may be susceptible to an attack.

In March 2007, the NRC issued 10 CFR 73.1, “Design Basis Threat,” to amend the requirements pertaining to design basis threats to include a cyber attack. Subsequently, in March 2009, the NRC issued 10 CFR 73.54, “Protection of Digital Computer and Communications Systems and Networks,” requiring licensees and combined license applicants to provide high assurance that nuclear power plant safety, security, and emergency preparedness functions are adequately protected from cyber attacks up to and including the design basis threat. The NRC also issued Regulatory Guide (RG) 5.71, “Cyber Security Programs for Nuclear Facilities,” which describes an acceptable method for complying with the Commission’s regulations regarding adequately protecting digital computers, communications systems, and networks associated with safety, security, and emergency preparedness functions of nuclear power plants from cyber attacks.

The new regulation required operating power reactor licensees to submit a cyber security plan, including an implementation schedule, to the NRC for review no later than November 23, 2009. All operating nuclear power plant licensees met the submission deadline. Applicants of combined licenses will submit cyber security plans as part of the process for acquiring a license. As a result of the amount of work and significant lead time required to fully implement all the provisions called for in licensee's NRC-approved cyber security plans, interim milestones were identified that emphasized completion of a set of prioritized activities by December 31, 2012.

The NRC developed an oversight program that includes cyber security inspector training, an inspection program, and a process for evaluating the significance of inspection findings. The inspection program includes developing TIs to be used in inspections for both the interim milestones and the full implementation of licensees' cyber security programs. This was accomplished collaboratively with stakeholders, including NRC staff and regional inspectors, members of industry, and representatives from the Department of Homeland Security, the Federal Energy Regulatory Commission, and the National Institute of Standards and Technology. Implementation of the interim milestones is currently being inspected.

In addition to ensuring the cyber security of nuclear power plants, the NRC is also responsible for securing its internal information technology systems and networks. Therefore, the NRC established the Computer Security Office in 2007 to address cyber security and information security issues within the agency.

Section 18.3.2.3 of this report discusses cyber security in more detail.

Digital Upgrades to Instrumentation and Control

The use of digital instrumentation and control raises issues that were not relevant to analog systems. Examples of such issues include:

- A common-cause failure attributable to software errors was not possible with analog systems. The potential for common-cause failure requires the consideration of diversity and defense-in-depth in the application of digital instrumentation and control systems.

- Interchannel communication, communication between nonsafety and safety systems, and system security and reliability must be reviewed closely to ensure that public safety is preserved.

- Highly integrated control room designs with safety and nonsafety displays and controls will be the norm for new reactor designs.

- Human factors design and quality assurance during all phases of software development, control, and validation and verification are critical.

The NRC's Digital Instrumentation and Control Steering Committee initiated task working groups to develop ISG documents for all high-priority technical issues associated with licensing digital instrumentation and control for nuclear power reactors. The working groups developed the ISG documents with significant input from external stakeholders through a series of public meetings and posted draft versions on the NRC Web site for public comment. The working groups addressed the following technical issues: (1) cyber security, (2) diversity and defense-in-depth,

(3) review of new reactor digital instrumentation and control probabilistic risk assessment (PRA), (4) highly-integrated control room communications, (5) highly-integrated control room human factors, 6) digital instrumentation and control safety system licensing processes, and (7) digital instrumentation and control in safety applications at fuel cycle facilities.

The NRC staff is using the guidance documents to conduct ongoing reviews. Feedback from licensees and NRC staff who have used the ISG documents has been positive. The staff used the ISGs in reviewing digital upgrades for the Wolf Creek and Oconee plants and in reviewing a number of design certifications and combined licenses for new plants.

The NRC is working to incorporate the ISGs into updated regulatory documents such as Standard Review Plans (SRPs), RGs, and NUREGs. The ISG for cyber security has been replaced by the new cyber security rule (i.e., 10 CFR 73.54) and its associated guidance. In light of these changes to the agency approach to cyber security, the staff has embarked on changes to clarify that cyber security will not be addressed as part of the 10 CFR Part 50 licensing reviews, but will be addressed as described in the previous section on Cyber Security. This approach will better enable the NRC and licensees to address an ever-evolving cyber threat over the operational lives of plants.

The Digital Instrumentation and Control Steering Committee completed its key objectives of improving the predictability and effectiveness of the digital instrumentation and control licensing process, and in December 2011, the Steering Committee was dissolved. The responsibilities for oversight of digital instrumentation and control licensing activities have been returned to the NRC staff. Engagement with the industry continues through periodic meetings on digital instrumentation and control issues and initiatives.

The results of the Digital Instrumentation and Control Steering Committee initiative included the publication of 7 ISG documents. The staff is working to incorporate the guidance of these publications into more permanent guidance or into industry standards. The last of these publications, DI&C-ISG-06, “Licensing Process,” issued in January 2011, is now being used to perform the review of the Diablo Canyon process protection system license amendment request. This licensing activity was approved as a pilot project for the implementation of DI&C-ISG-06 in accordance with 10 CFR Part 170.11(b).

The NRC staff has approved five digital platform topical reports to support future digital instrumentation and control licensing submittals and is reviewing three additional digital platform topical reports. The NRC staff is providing additional guidance to address the appropriate implementation of digital modifications in nuclear power plants under 10 CFR 50.59, “Changes, Tests, and Experiments.”

Additionally, the staff is preparing a revision to the existing regulations pertaining to digital safety systems to incorporate by reference the latest version of Institute of Electrical and Electronics Engineers (IEEE) Standard 603-2009, “Criteria for Safety Systems for Nuclear Power Generating Stations.”

Article 18 of this report discusses the digital instrumentation and control in new plant designs.

Moisture Effects on Underground Cables

The NRC began a detailed review of underground electrical power cables after an increasing trend in moisture-induced cable failures was identified. The failed cables had been exposed to condensation, wetting, submergence, and other environmental stresses that resulted in insulation degradation. Since most of the cables exposed to this environment were not designed for continuous wetting or submergence, there is an increasing possibility of multiple failures, which in turn could initiate a plant shutdown or disable accident mitigation systems.

On February 7, 2007, the NRC issued GL 2007-01, "Inaccessible or Underground Power Cable Failures That Disable Accident Mitigation Systems or Cause Plant Transients," to inform licensees that the failure of certain power cables can affect the functionality of multiple accident mitigation systems or cause plant transients. The NRC asked the licensees to provide information on inaccessible or underground power cable failures for all cables within the scope of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (the Maintenance Rule).

Based on the review of licensee's responses to GL 2007-01, the NRC staff identified 269 cable failures at U.S. nuclear power plants. Licensees applying for a 20-year license renewal, in accordance with 10 CFR Part 54 requirements, have agreed to implement a cable testing program during the period of extended operation for a limited number of cables that are within the scope of licensee renewal, but only a few plants have established a cable testing program for the current operating period. Data obtained from the responses to GL 2007-01 show an increasing trend in cable failures within the plants' current 40-year licensing period of operation. The predominant factor contributing to cable failures at nuclear power plants appears to be the presence of water or moisture resulting in intrusion, because of the submergence of underground cables in water. If cables have been exposed to conditions for which they are not designed, licensees need to demonstrate, through adequate testing, that there is reasonable assurance that the cables can perform their intended design function. Licensees should also minimize the amount of moisture in underground cable raceways, conduits, and cable vaults. Cables not designed or qualified for, but exposed to, wet or submerged environments have the potential to degrade. Cable degradation increases the probability that more than one cable will fail on demand because of a cable fault, lightning surge, or a switching transient. Although a single failure is within the plant design basis, multiple failures of this kind would be challenging for plant operators. Also, an increased potential exists for a common-mode failure of accident mitigating system cables if they are subjected to the same environment and degradation mechanism for which they are not designed or qualified.

NRC regulations in 10 CFR Part 50 require licensees to assess the condition of systems and components in a manner sufficient to provide reasonable assurance that they are capable of fulfilling their intended functions, and that a test program to ensure that components will perform satisfactorily in service is identified and performed. Licensees should have a program for using available diagnostic cable testing methods to assess cable condition to ensure the insulation is not degraded over the life of the plant. The NRC Reactor Oversight Process baseline inspection procedures have been revised to inspect the licensees' cable performance monitoring activities. To date, NRC inspectors have identified various violations of 10 CFR Part 50 requirements at several facilities.

In January 2010, the NRC issued NUREG/CR-7000, "Essential Elements of an Electric Cable Condition Monitoring Program," to inform licensees of the types of cable testing methods currently available to detect cable insulation degradation. In addition, EPRI has developed a model cable monitoring program to provide licensees with information on creating such a program. On December 2, 2010, the NRC issued IN 2010-02, “Submerged Electrical Cables,” to inform licensees and combined license applicants of observations of protracted cable submergence in water, recent NRC inspection findings, and the results of licensees’ responses to GL 2007-01.

In April 2012, the NRC issued RG 1.218, “Condition Monitoring Techniques for Electric Cables Used in Nuclear Power Plants.” This guide describes a method and techniques that the NRC staff considers acceptable for use in implementing the regulatory requirements for monitoring the performance of electric cables used in nuclear power plants.

Containment Pressure Credit for Emergency Core Cooling System Pump Net Positive Suction Head

NRC RG 1.1, “Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps,” dated November 2, 1970, states that the pressure in containment before the postulated accident should be used when determining the available net positive suction head of emergency core cooling system and containment heat removal system pumps. Before the NRC issued this guidance document, some reactors were designed and licensed using the calculated containment accident pressure in calculating the net positive suction head margin.

The agency modified this guidance in RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss of Coolant Accident,” Revision 3, issued November 2003, which permitted certain operating reactors to use containment accident pressure when modification of the reactor design was impractical. The modification to the guidance of RG 1.1 recognized the fact that in certain cases it was not practical to avoid using containment accident pressure. Such cases included subatmospheric containments, application of a larger debris source term following a loss-of-coolant accident, and an increase in licensed thermal power (or power uprates).

As a result of discussions with the NRC Advisory Committee on Reactor Safeguards, the staff has reexamined this issue with the goal of studying the related pump phenomena and quantifying margins both in terms of pump cavitation limits and containment accident response. Some of the subjects examined include the effect of containment integrity testing frequency on failure probabilities, the uncertainty in required net positive suction head, assessment of cavitation erosion of the pump impeller, and the mechanical performance of centrifugal pumps with various degrees of cavitation.

The staff developed draft deterministic guidance considering the Advisory Committee on Reactor Safeguards recommendations for quantifying uncertainty in the net positive suction head calculations. The Commission directed the staff to conduct the net positive suction head analysis crediting containment accident pressure reviews consistent with current staff practice but also to implement the new staff deterministic guidance. The Commission directed the staff to assure that the defense-in-depth philosophy is interpreted and implemented consistently in RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 2, issued May 2011. The draft guidance was sent to the industry for comments and for proposing implementation methods if containment accident pressure is credited for calculating net positive suction head margin. The staff has discussed the proposed draft guidance with the Boiling Water Reactor Owners

Group and the Pressurized Water Reactor Owners Group. ISG will be developed after completing the review of a pilot application for an extended power uprate.

Gas Voiding Issues in Light-Water Reactor Safety Systems

The accumulation of gas in systems that are important to safety has been a continuing problem since the first light-water nuclear power plants were placed into operation. Early manifestations of the issue included pipe hanger damage as a result of water hammer in residual heat removal systems when the systems were started and the loss of residual heat removal when the pumps became gas-bound. This led to recognition of potential problems with the emergency core cooling systems since much of the residual heat removal system also serves as the low-pressure – high flow rate portion of the emergency core cooling system, and similar problems could occur in the low-pressure and high-pressure emergency core cooling systems if they were placed in operation in response to a loss-of-coolant accident. Consequently, numerous publications were issued to address the issue, technical specifications were developed to require pump discharge piping to be full of water to address the water hammer issue, and steps were taken to prevent gas ingestion into pumps. Before 2008, the actions were not fully successful because of a failure to understand the root causes of gas accumulation and to address comprehensively the potentially affected systems and the phenomena associated with gas accumulation and movement before, during, and after system startup.

The root causes of gas accumulation include: (1) designs that allow gas introduction and accumulation, (2) licensees failing to properly fill and vent the system following drain-down or maintenance, (3) ineffective gas accumulation controls during operation, (4) inappropriate technical specifications regarding the scope and frequency of inspections for gas accumulation, and (5) unanticipated problems with keep-full systems.

GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems," dated January 11, 2008, addressed the issue for several important safety systems through indepth coverage of the phenomena and the operating processes necessary to prevent event occurrence as a result of gas. The U.S. nuclear industry provided a detailed response to GL 2008-01 that included: (1) suction pipe testing, (2) development of analysis methodologies, (3) system walkdowns, including precision measurement of piping configurations, (4) void measurements using ultra-sonics, rewritten and new procedures, (5) extensive operator training, and (6) hardware changes such as the addition of vent valves and tanks to remove gas from piping before it becomes a concern. These followup actions have resulted in an enhanced understanding of the issues and implementation of measures to minimize future problems. As a result, there is an increased confidence that the systems will perform their safety-related functions when required to do so. Further improvements are underway. These include the development of improved void behavior analysis methods, increased indepth coverage of transient behavior during pump starts, improved technical specification coverage including surveillance requirements, increased technical coverage including systems that were not identified in GL 2008-01, and improvements in plant operation, including areas such as the corrective action plan, procedures, and operator training.

The NRC followed up on the industry activities by reviewing licensee responses to GL 2008-01 and performing inspections at the 104 nuclear power plants licensed in the United States. The scope of these activities is illustrated by the generic review instructions the NRC uses in providing inspection suggestions to its inspectors in accordance with TI 2515/177, "Managing

Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)," dated June 9, 2009, and in the July 5, 2011, memorandum that documented the completion of the staff's review of GL 2008-01. The scope of industry participation is evident in the continuing support of technical specification improvements and in the release of Nuclear Energy Institute (NEI) guidance document NEI 09-10, "Guidelines for Effective Prevention and Management of System Gas Accumulation," Revision 1a, dated November 1, 2012.

Enhancements to Emergency Preparedness Regulations

The final rulemaking regarding emergency preparedness at U.S. nuclear power plants was issued in late 2011. The implementation of the rule is described in Section 1.3.3 of this report.

The new requirements should enhance the licensees' ability to prepare and implement certain emergency preparedness and protective measures in the event of a radiological emergency. These changes also address, in part, security issues identified after the 2001 terrorist events; clarify regulations to achieve consistent emergency plan implementation among licensees; and modify certain emergency preparedness requirements to be more effective and efficient.

1.3.2 Current Safety and Regulatory Issues

The NRC and its licensees are currently working with the following potential safety and regulatory issues:

- Fukushima lessons learned
- steam generator integrity
- nondestructive evaluations
- concrete structural issues
- cumulative effects of regulation
- evaluation of economic consequences
- counterfeit, fraudulent, and suspect items
- construction inspection program lessons learned

Fukushima Lessons Learned

Immediately following the accident at Fukushima, the NRC took actions to ensure that there were no immediate safety concerns at U.S. facilities, and actions that verified nuclear power plant operators' preparedness to respond to and mitigate the consequences of beyond-design-basis events. These actions included issuance of IN 2011-05, "Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants," dated March 18, 2011, to inform U.S. licensees regarding what was known about the Fukushima accident. The NRC also issued two inspection procedures, called temporary instructions, to NRC regional and/or resident inspectors to evaluate specific aspects of licensee preparedness to respond to an event like that which occurred at the Fukushima facility (TI 2515/183, "Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event," dated March 23, 2011, and TI 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)," dated April 29, 2011). Finally, the NRC issued Bulletin 2011-01, "Mitigating Strategies," dated May 11, 2011, to request information from U.S. licensees regarding their preparations for dealing with such an event.

On March 23, 2011, the Commission approved formation of a Near-Term Task Force (NTTF), comprised of senior NRC staff and management, to systematically and methodically review the NRC's processes and regulations in light of the Fukushima accident. The Commission tasked the NTTF with determining whether the NRC should make additional improvements to its regulatory system, and to make policy recommendations to the Commission. The NTTF issued its report, entitled "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of the Insights from the Fukushima Daiichi Accident," on July 12, 2011. The NTTF concluded that continued operation of U.S. nuclear plants and ongoing NRC licensing activities posed no imminent risk. The NTTF also concluded that enhancements to safety and emergency preparedness are warranted, and made 12 overarching recommendations for Commission consideration, including the establishment of a logical, systematic, and coherent regulatory framework that appropriately balances multiple layers of protection and risk considerations to deal with events beyond the current NRC design basis (Recommendation 1).

The NRC formed the Japan Lessons-Learned Project Directorate to perform a longer-term review of the March 11, 2011, Japanese earthquake and tsunami. The Japan Lessons-Learned Project Directorate reports to a Steering Committee of senior NRC officials, which is chaired by the Deputy Executive Director for Reactor and Preparedness Programs and is composed of Office Directors from many of the NRC program offices and regional offices. The Japan Lessons-Learned Project Directorate prioritized the NTTF recommendations by tiers and expanded upon the task force recommendations to include ideas from the international community, the U.S. Congress, and other stakeholders, and continues to evaluate these additional recommendations, as appropriate.

<u>Tier 1 Recommendations</u>

The first tier consists of actions that could begin without unnecessary delay. To determine and recommend near-term regulatory actions that should be initiated without delay, the staff considered whether any of the recommendations identified an imminent hazard to public health and safety. The staff concluded that none of the recommendations rose to this level. The staff then identified, under the Tier 1 activities, a subset of actions that has the potential to provide significant safety improvement in the near-term.

- seismic and flood hazard reevaluations (*Recommendation 2.1*)
- seismic and flood walkdowns (*Recommendation 2.3*)
- station blackout (SBO) regulatory actions (R*ecommendation 4.1*)
- mitigating strategies for beyond-design-basis events (*Recommendation 4.2*)
- reliable hardened vents for Mark I and Mark II containments (R*ecommendation 5.1*)
- SFP instrumentation (*Recommendation 7.1*)
- strengthening and integration of emergency operating procedures, severe accident management guidelines (SAMGs), and extensive damage mitigation guidelines (*Recommendation 8*)
- emergency preparedness regulatory actions (staffing and communications) (*Recommendation 9.3*)

Tier 2 Recommendations

The second tier recommendations are actions that originally could not be initiated because of a need for further technical assessment and alignment, dependence on Tier 1 issues, or lack of availability of critical skill sets. These actions do not require long-term study and can be initiated when sufficient technical information and applicable resources become available.

- SFP makeup capability (*Recommendation 7.2, 7.3, 7.4, and 7.5*)
- emergency preparedness actions (*Recommendation 9.3*)
- other external hazards reevaluation (e.g., tornados, hurricanes, and drought)

As a result of further assessment after these items were prioritized as Tier 2, the NRC determined that SFP makeup capability and the emergency preparedness actions could be adequately addressed as part of the Tier 1 activity for mitigating strategies for beyond-design-basis events. Therefore, these two Tier 2 activities have been consolidated into Tier 1.

Tier 3 Recommendations

The third tier consists of actions that require further staff study to support regulatory action; need the result of an associated short-term action to inform the long term action; depend on the availability of critical skill sets; or relate to potential revisions to the regulatory framework that balances defense-in-depth and risk considerations. The staff has focused its initial efforts on developing the schedules, milestones, and resources associated with Tier 1 and Tier 2 activities. Once the staff has completed its evaluation of the resource impacts of the Tier 1 and Tier 2 recommendations, it will be able to more accurately address the Tier 3 recommendations.

- 10-year confirmation of seismic and flooding hazards (dependent on Recommendation 2.1) (*Recommendation 2.2*)
- potential enhancements to the capability to prevent or mitigate seismically-induced fires and floods (long-term evaluation) (*Recommendation 3*)
- reliable hardened vents for other containment designs (long-term evaluation) (*Recommendation 5.2*)
- Hydrogen control and mitigation inside containment or in other buildings (long-term evaluation) (*Recommendation 6*)
- emergency preparedness enhancements for prolonged SBO and multi-unit events (dependent on availability of critical skill sets) (*Recommendation 9.1/9.2*)
- Emergency Response Data System capability (related to long-term evaluation Recommendation 10) (*Recommendation 9.3*)
- additional emergency preparedness topics for prolonged SBO and multi-unit events (long-term evaluation) (*Recommendation 10*)
- emergency preparedness topics for decisionmaking, radiation monitoring, and public education (long-term evaluation) (*Recommendation 11*)
- Reactor Oversight Process modifications to reflect the recommended defense-in-depth framework (dependent on Recommendation 1) (*Recommendation 12.1*)
- staff training on severe accidents and resident inspector training on severe accident management guidelines (dependent on Recommendation 8) (*Recommendation 12.2*)
- basis of emergency planning zone size (*additional issue*)
- prestaging of potassium iodide beyond 10 miles (*additional issue*)

- transfer of spent fuel to dry cask storage (*additional issue*)

Using existing regulatory processes (e.g., orders, rulemaking, and requests for information (RFI)), the Japan Lessons-Learned Project Directorate provides project management and oversight of implementation of Fukushima lessons learned.

The NRC issued the first regulatory requirements, in the form of orders, for the Nation's 104 operating reactors based on lessons learned at Fukushima. These orders require safety enhancements of operating reactors, construction permit holders, and combined license holders. Specifically, they require nuclear power plants to implement safety enhancements related to (1) mitigation strategies to respond to external events resulting in the loss of power at plants, (2) ensuring reliable hardened containment vents for BWR Mark I and II designs, and (3) enhancing SFP instrumentation.

The NRC issued an RFI requiring each reactor to reevaluate the seismic and flooding hazards at its site using present-day methods and information, conduct walkdowns of its facilities to ensure protection against the hazards in its current design-basis, and assess its emergency communications systems and staffing levels. Licensees began submitting responses early in 2013 and the NRC staff is continuing to evaluate the responses.

In June 2011, the NRC issued a Statement of Policy to set forth its expectation for individuals and organizations performing or overseeing regulated activities to establish and maintain a positive safety culture commensurate with the safety and security significance of their activities and the nature and complexity of their organizations and functions. U.S. nuclear licensees have responded and continue to respond openly to NRC requests and initiatives that confirm and ensure adequate measures to protect public health and safety considering the lessons learned following the Fukushima nuclear accident.

Details about the NRC activities associated with the Fukushima lessons learned are specified within the individual Articles of this report and in NUREG-1650, "The United States of America National Report for the 2012 Convention on Nuclear Safety Extraordinary Meeting," Revision 4, issued July 2012. Specifically, per paragraph 23 of the Final Summary Report for the 2nd CNS Extraordinary Meeting, the Contracting Parties agreed that in particular the National Reports should cover:

(a) The results of reassessments of external events, of periodic safety assessments and of any peer reviews, and any follow-up actions taken or planned, including upgrading measures.

    The NRC is undertaking near-term regulatory activities to reevaluate and enhance, as necessary, the protection of SSCs against design-basis seismic and flooding events for all operating reactors in the U.S. These activities are based on NTTF Recommendations 2.1 and 2.3, as modified by subsequent NRC management direction. These activities include requesting that licensees reevaluate the seismic and flooding hazards at their sites using updated methods, and perform "walkdowns" to identify plant-specific vulnerabilities. Additional details and actions for other external events are discussed in Section 18.5.1 of this report.

(b) Actions taken or planned to cope with natural hazards more severe than those considered in the design basis.

The NRC is evaluating topics related to external events beyond the design-basis. Activities associated to these topics include the following:

- a rulemaking to require licensees to confirm seismic and flooding hazards every 10 years and address any new and significant information
- potential enhancements to licensees' capability to prevent or mitigate seismically induced fires and floods

Additional details are discussed in Section 18.5.3 of this report.

(c) For new nuclear power plants, improved safety features and additional improvements, if any, to address external hazards and to prevent accidents and, should an accident occur, to mitigate its effects and avoid off-site contamination.

In response to the Fukushima accident, the NRC used its regulatory processes to request that licensees reevaluate the seismic and flooding hazard at their sites using present-day regulatory guidance and methodologies and, if necessary, to perform a risk evaluation. Information from these evaluations will be used to determine whether additional regulatory actions are necessary (e.g., update the design-basis and SSCs important to safety) to protect against the updated understanding of hazards. Additional information is discussed in Articles 17 and 18 of this report.

(d) Upgrading of accident management measures for extreme natural events, including, for example, measures to ensure core cooling and SFP cooling, the provision of alternate water sources for the reactor and for the SFP, the availability of the electrical power supply, measures to ensure containment integrity, and filtration strategies and hydrogen management for the containment; the development of probabilistic safety assessments to identify additional accident management measures should be considered as a possible future activity.

The NTTF recommendations for upgrading accident management measures were discussed earlier in this section (i.e., Section1.3.2) of this report. The development of probabilistic safety assessments to identify additional accident management measures are discussed in Section 18.5.2 of this report.

(e) Measures taken or planned to ensure the effective independence of the regulatory body from undue influence, including, where appropriate, information on the hosting of IRRS missions.

As noted in Section 8.3 of this report, the U.S. Congress created the NRC as an independent agency in 1975. As a result of the Fukushima nuclear accident, there have been no changes in the U.S. legislative framework that governs the NRC and the regulations of the U.S. nuclear industry. Regarding IAEA peer review missions, the U.S. hosted an IRRS mission in 2010 and has scheduled its followup mission for early 2014. Additional details are discussed in section 8.1.5.2 of this report.

(f) Enhancements of emergency preparedness and response measures, including, for example, for multi-unit sites, approaches and methods of source term estimation and initiatives in the field of remediation. The enhancements should include defining the additional responsibilities up to appropriate levels of the national government and the development of procedures and joint actions of various agencies and improvements in international cooperation.

Following the Fukushima event, the NRC undertook actions to enhance emergency preparedness with respect to communications and staffing given a multi-unit event and a prolonged SBO. The accident highlighted the need to identify the staff needed to respond to a multi-unit event given a prolonged SBO. Additional information can be found in Section 16.9 of this report.

(g) Information on how IAEA safety standards are taken into account.

The NRC actively participates in the development of the IAEA's safety standards. Where appropriate, the NRC also references the safety standards in NRC regulations. Additional information can be found in Section 8.1.5.1 of this report.

(h) Information on activities undertaken to enhance openness and transparency for all stakeholders.

Openness is the second of six "Principles of Good Regulation" that the NRC first established in 1977. These principles guide all of the agency's activities. Openness is also one of seven organizational values, adopted in 1995, to which the agency adheres in all its work. After the Fukushima event, the NRC updated its crisis communication plan with lessons learned and added staff to the Office of Public Affairs' technical briefer list to support public and media outreach efforts in future events. Additional information can be found in Sections 8.1.7 and 8.3 of this report.

Steam Generator Integrity

Steam generators in PWRs contain components that form part of the reactor coolant pressure boundary (e.g., tubing and the channel head). Managing steam generator tube degradation has been a significant area of focus by industry since the first operating reactors were brought into service. The industry has moved to different heat treatments and alloys in successive generations of steam generators, in an effort to decrease the susceptibility of steam generator tubing to various corrosion mechanisms. Operating conditions and maintenance items can potentially affect the useful lifetime of a steam generator and may affect the integrity of the steam generator tubing. Several examples of steam generator integrity issues with implications for the NRC staff are provided below.

<u>Tube-to-Tube Wear</u>

Wear attributed to tube-to-tube contact has been detected in both once-through and recirculating steam generators.

Replacement Once-Through Steam Generators

Wear indications, attributed to tube-to-tube contact, at Three Mile Island Unit 1 were first reported in fall 2011, after one cycle of operation with the replacement steam generators. After the Three Mile Island findings were shared with other plants with once-through steam generators, subsequent re-analysis of prior eddy current inspection data by these plants indicated tube-to-tube wear was present at some of the other units. The re-analyses indicated that the tube-to-tube wear at these plants is shallow and slow growing. Subsequent to these findings, NRC issued IN 2012-07, "Tube-to-Tube Contact Resulting in Wear in Once-Through Steam Generators," dated July 17, 2012, to provide licensees with lessons learned from the discovery of these indications. Licensees were expected to review the information for applicability and consider actions, as appropriate, to avoid similar problems. These findings highlight the importance of performing comprehensive inspections of new and replacement equipment to ensure they perform as expected. The cause of the tube-to-tube contact is currently being evaluated. The NRC staff will review the results of this evaluation to determine if any further regulatory action is needed.

Replacement Recirculating Steam Generators

San Onofre Nuclear Generating Station replaced the Unit 2 steam generators in 2010 and the Unit 3 steam generators in 2011. On January 31, 2012, San Onofre Unit 3 was operating at 100 percent rated thermal power when a primary-to-secondary leak was detected. Although the leak rate was initially small, it increased enough in a short period of time that the plant was shut down. Unit 3 was in its first cycle of operation with replacement steam generators.

At the time of the leak in Unit 3, Unit 2 already was shut down for maintenance and refueling, having just completed its first cycle of operation with replacement steam generators. Tube wear was detected at a number of locations in both units. The wear was attributed to the tubes interacting with tube support plates, antivibration bars, retainer bars, and other tubes. The wear attributed to the retainer bars and tube-to-tube contact was not expected. All tubes in Unit 2 had adequate integrity. At Unit 3, there were eight tubes that did not have adequate integrity because of tube-to-tube wear.

A root cause evaluation report that the plant owner prepared indicated that the U-bend portion of some of the tubes experienced fluid elastic instability in the in-plane direction which caused the tubes to wear against each other. The wear in the tubes near the retainer bars was a result of the design of the smaller diameter retainer bars, which was insufficient to prevent excessive flow-induced vibration of the retainer bar.

By letter dated June 12, 2013, Southern California Edison (i.e., the licensee) notified the NRC of its decision to permanently cease operations at San Onofre Nuclear Generating Station Units 2 and 3. Additional information on the decision to cease operations at San Onofre is discussed in Section 6.2 of this report.

<u>Wear Other than Tube-to-Tube Wear</u>

Several units with replacement steam generators have detected many indications of tube wear, including Saint Lucie Unit 2 and Oconee Units 1, 2, and 3.

Saint Lucie Unit 2 has two recirculating steam generators. During the first refueling outage after steam generator replacement in spring 2009, approximately 5,800 tube wear indications were detected. Although some wear was reported at tube support plate elevations and at another location, most of the wear was at the antivibration bars. A root cause analysis performed by the plant revealed that some of the antivibration bars were out of position by a few mils (thousandths of an inch), as a result of plastic deformation that occurred during fabrication.

Widespread tubing wear at tube support plate locations on the steam generators at Oconee has been observed at all three units with the most probable cause being the precise alignment of the tube supports that allows small excitation forces to cause tube vibration since the tubes are minimally damped and consequently very responsive over their lengths.

Although a large number of wear indications have been detected, no loss of tube integrity has occurred at Saint Lucie Unit 2 or the Oconee units as a result of these wear indications.

No additional regulatory action has been deemed necessary at this time; however, the NRC staff continues to monitor the results of the steam generator tube inspections at these units.

Low Alloy Steel Channel Head Corrosion Operating Experience

In response to international operating experience on corrosion of the low-alloy steel steam generator channel head beneath the channel head cladding (in the vicinity of the channel head drain line), some U.S. plants have performed inspections of their steam generator channel heads. As of April 2013, no corrosion near the channel head drain line has been identified in the U.S. steam generators; however, some minor corrosion of the low alloy steel channel head at a different location was observed at one U.S. facility. The NRC staff is currently developing an Information Notice on this topic.

Nondestructive Evaluations

There have been several recent issues of operating experience where nondestructive evaluation has yielded results that have caused the NRC to take action. In some cases, nondestructive evaluation results have identified the occurrence of degradation at rates that are different than had been anticipated causing the NRC to review inspection requirements. Also, some unexpected evaluation results have caused the NRC to reevaluate its regulatory practices. In other cases, the failure of nondestructive evaluation to find significant degradation has led to adaptive approaches to improve the performance of the evaluation. Several examples of the implications of these nondestructive evaluation issues are provided below.

Failed Inspection of North Anna 1 Weld

In March 2012, a manual ultrasonic testing examination of a steam generator inlet dissimilar metal weld at the North Anna Unit 1 power plant failed to detect five axial flaws ranging from 40 percent to 80 percent through-wall. The failure of an inspection to detect five deep flaws has caused a series of responses, both at the NRC and in industry. The NRC staff raised concerns about the inspection procedure and the qualification process for nondestructive evaluation procedures, and is regularly meeting with industry to improve inspections to prevent similar occurrences in the future.

The licensee was preparing to apply a full structural overlay to the steam generator inlet dissimilar metal weld. In accordance with the ASME Code requirements, before applying the overlay, the licensee needed to perform nondestructive examination of the weld to assess its pre-overlay condition. Both the overlaying process and the inspection were complicated by the design of the weld, which had a taper that made the application of the weld overlay very challenging. As such, the taper was machined off prior to overlay applications. After the machining process was complete, two leaking axial cracks became evident. A followup inspection on the now-flat weld found the presence of three more axial cracks, each roughly 2.5 inches deep. These cracks were determined to be primary water stress-corrosion cracks.

NRC staff, with assistance from the Pacific Northwest National Laboratory, conducted an independent review of the circumstances that led to the failure to detect these five primary water stress-corrosion cracks. PNNL-21546, “Evaluation of Manual Ultrasonic Examinations Applied to Detect Flaws in Primary System Dissimilar Metal Welds at North Anna Power Station,” issued June 2012, describes the NRC’s review and concerns. It was determined that the licensee used ineffective protocols and qualification of procedures under the auspices of the Performance Demonstration Initiative. The NRC staff is working with the Performance Demonstration Initiative to improve relevant processes to prevent a similarly ineffective procedure from being used in the future.

As a result of the North Anna event, broader issues with respect to the effectiveness and reliability of nondestructive examinations have been identified. As a result, industry responded by forming the Nondestructive Evaluation Improvement Focus Group, which is evaluating many nondestructive evaluation issues and developing recommendations to improve future evaluations. There has been an ongoing dialogue between the NRC staff and the focus group on these subjects, including 6 public meetings. These discussions covered a broad range of topics related to dissimilar metal weld examinations, and specific to what was learned as a result of the North Anna event, improvements that can be made to the site-specific process and the regulatory requirements for changing essential variables in qualified examination procedures.

Indications in the Belgian Pressure Vessel Forgings

In June 2012, a new ultrasonic inspection was performed to examine the pressure vessel for possible underclad cracks at the Doel Unit 3 plant in Belgium. Although no underclad cracks were detected, the inspection detected nearly laminar indications in the lower and upper shells of the vessel. As this inspection was focused near the inside diameter of the vessel, a followup inspection was performed in July 2012, with an array of ultrasonic transducers to inspect the full volume of the vessel forging rings. This followup inspection found more than 8,000 nearly laminar indications in the Doel 3 pressure vessel forging rings, with a typical size of roughly 10 millimeters in diameter. In September 2012, the pressure vessel of the Belgian Tihange Unit 2 plant, which was made using forgings from the same manufacturing facility, also was inspected. The inspection of the Tihange Unit 2 forging rings found over 2,000 similar indications in the lower and upper shells. The indications in Doel Unit 3 and Tihange Unit 2 were determined to be hydrogen flakes introduced during the vessel manufacture.

Preliminary fracture mechanics calculations show that the laminar orientation of the flaws makes them relatively benign to the toughness of the pressure vessel. The Belgian regulator, the Federal Agency for Nuclear Control, has used these preliminary fracture mechanics calculations to determine that Doel Unit 3 and Tihange Unit 2 do not, as of yet, need to be permanently shut down. The Federal Agency for Nuclear Control has, however, placed a large number of conditions on the restart of both reactors. Among these conditions, the licensee will need to perform a large experimental evaluation on the effects of hydrogen flakes on the fracture resistance of the material and use this information in a detailed analysis of the effects of the hydrogen flakes on the structural integrity of the pressure vessels.

The NRC staff reviewed fabrication information and determined that several reactor pressure vessels in the United States contain ring forgings that had been produced in the same fabrication shop as the rings in the Doel Unit 3 and Tihange Unit 2 vessels. The NRC also determined that other vessels in the United States contained ring forgings that had been produced in other fabrication shops. The staff notified industry about the possibility of hydrogen flaking in reactor pressure vessel ring forgings. In response, the industry indicated it was planning to retrieve the original fabrication records of ultrasonic examinations of vessel ring forgings. The industry also intends to document the ability of the construction-era ultrasonic examination techniques to detect indications similar in nature to hydrogen flakes, and to document the requirement for recording such indications. Subsequently, the industry will review the records to determine whether there is any evidence that any vessel forgings in the United States are affected. Independent of the records review effort, the industry will perform bounding structural integrity assessments to ensure that reactor pressure vessel integrity will be maintained under all conditions, even in the presence of hydrogen flaking.
The NRC staff is finalizing an IN to ensure that all licensees are aware of the Belgium findings.

The NRC hosted a public meeting to discuss the potential for, and implications of, hydrogen flaking in forgings on March 5, 2013. The NRC is planning to host a second public meeting where industry will present the finding of their investigations and analyses of the U.S. vessels. The staff is continuing to monitor industry progress on records reviews and structural integrity assessments. Should additional information become available that suggests that hydrogen flaking may be present or may be a structural integrity challenge to the operation of a U.S. reactor pressure vessel, the NRC will take appropriate action. The staff's current understanding is that the identified hydrogen flaking is structurally insignificant. Accordingly, there are no current plans to require additional ultrasonic examinations to look for hydrogen flaking.

In the longer term, the staff and industry have agreed to approach the ASME to ensure that lessons learned from the discovery of hydrogen flakes in an operating reactor pressure vessel are appropriately incorporated into applicable codes and standards.

<u>Identification of Cracking in a Bottom Mounted Instrument Nozzle at a French Plant</u>

During an inspection of a bottom mounted instrument nozzle at a plant in France, ultrasonic nondestructive evaluation identified cracks in the nozzle material adjacent to the J-groove weld that attaches the nozzle to the bottom head. The French operational inspection agency, the Nuclear Safety Authority, met with NRC counterparts and described the findings and the French regulatory response to require ultrasonic inspection of bottom mounted nozzles at all reactors. During the information exchange meeting, the NRC staff provided information related to similar findings of cracking of a bottom mounted instrument nozzle at South Texas Project Nuclear Plant in 2003. Current requirements to inspect the bottom mounted instrumentation nozzles at U.S. plants are contained in the ASME Code. The Code currently only requires visual examination of the outer surface of the reactor vessel where the bottom mounted nozzles exit the lower head. As a result of the nondestructive evaluation findings at the French plant, the NRC staff approached the ASME to initiate Code changes to require volumetric inservice inspection of bottom mounted instrument nozzles for all nuclear plants having material susceptible to primary water stress-corrosion cracking. The Code changes are currently under development.

Concrete Structural Issues

Since 2009, several significant conditions adverse to quality have occurred or were discovered in safety-related concrete structures of operating reactors in the United States. These conditions involve the following:

- containment delamination at Crystal River Unit 3
- shield building laminar cracking at Davis Besse
- alkali-silica reaction concrete degradation at Seabrook

Each of the above issues was or is being addressed by the respective licensee under its Corrective Action Program. A brief description of each of these issues is provided below:

<u>Containment Delamination Issue at Crystal River Unit 3</u>

During the October 2009, Unit 3 refueling outage, while cutting a construction opening by hydrodemolition for a steam generator replacement project, the licensee discovered a delamination of the cylindrical wall of the post-tensioned concrete containment. The licensee's condition assessment determined that the extent of the delamination was limited to bay 3-4, and corresponded to an hourglass shaped area including the steam generator replacement construction opening. The containment in Crystal River Unit 3 had experienced a previous delamination in the dome area during construction in the mid-1970s. This area was repaired and a structural integrity test performed before the plant was made operational.

The delaminated condition was not an immediate safety concern, since the plant was shut down when the condition was discovered and has remained in a safe shutdown condition since discovery. As a result of the event, the NRC conducted a special inspection and held several public meetings.

The licensee's completed investigation of the delaminated condition included four major areas: (1) condition assessment to determine the extent of condition; (2) root cause analysis; (3) design-basis analysis; and (4) repair analysis and design of repair implementation.

Along with other contributing causal factors, the technical root cause was determined to be inadequate scope and sequence of detensioning of tendons associated with the steam generator replacement construction opening activities, which resulted in redistributed stresses that exceeded the tensile capacity of the concrete. Other contributing factors included containment concrete with lower than normal tensile strength and limited crack-arresting capability because of the fragility of the soft coarse aggregate used. Through state-of-the-art computer models, the licensee determined that none of the individual contributing factors, on their own, would have caused a delamination. Rather, the complex interplay between all the contributing causes resulted in the delamination, with the driver being tendon detensioning.

The licensee removed and replaced the concrete in the delaminated area. Following additional detensioning for repair, new through-wall vertical cracks were discovered in all six bays. The licensee evaluated the effect of these cracks on the design-basis. Repair activities were initiated.

On March 14, 2011, during the final phase of containment retensioning to complete the repair of bay 3-4, the Unit 3 containment wall experienced a second delamination of concrete in bay 5-6, an isolated area of surface spalling, and a third delamination within bay 1-2. In June 2011, the licensee announced a tentative plan to repair the delamination(s) in the Unit 3 containment building. The plan posed significant construction challenges since the containment bays that needed to be repaired were surrounded by rooflines, and in some areas inside, obstructed by adjacent buildings. The proposed repair was estimated to take at least 5 years to complete and would involve significant construction and replacement power costs. In February 2013, the licensee decided to terminate the proposed repair of the containment and announced that it would permanently cease operations at Crystal River Unit 3.

Additional information on the decision to cease operations in Crystal River Unit 3 is discussed in Section 6.2.

Shield Building Laminar Cracking at Davis-Besse

During an October 2011, mid-cycle outage, while cutting a construction opening to replace the reactor vessel closure head, laminar cracking was identified in the architectural flute shoulder area of the shield building cylindrical wall at the Davis-Besse nuclear power plant. The shield building is a reinforced concrete structure that surrounds the free-standing steel containment vessel. The building provides a biological shield and environmental protection during accidents.

The licensee performed extensive nondestructive testing, which confirmed cracking in 15 flute shoulder regions. The cracks were very tight (i.e., hairline cracks) and the rebar and concrete were generally found to be in good condition. The licensee determined that a blizzard in 1978 resulted in environmental conditions during construction that enabled moisture to penetrate the shield building concrete, freeze, and expand. This created stresses and initiated the subsurface laminar cracking. The root cause for the concrete laminar cracking was a design specification for construction of the shield building, which did not require application of an exterior moisture barrier.

The licensee performed structural evaluations to capture bounding conditions and took corrective actions including (1) establishment of a test program to investigate the steel reinforcement capacity, (2) development of an engineering plan to reestablish the design and licensing bases for the shield building, (3) development of a procedure for long-term monitoring, and (4) installment of a sealant system.

On December 2, 2011, the NRC issued a letter to the licensee regarding continued operability of the shield building at Davis-Besse and license commitments. The licensee completed all commitment actions by summer 2012. The NRC staff has provided extensive and rigorous oversight of the licensee’s testing and evaluations of this issue.

The licensee projects completion of the required analyses to reconcile the shield building design and the current licensing basis by the end of 2013. The licensee will continue to obtain and test core bore samples and inspect the shield building to ensure no changes in the cracking. Even though the licensee performed extensive review and testing for the cracking in the shield building, plant restart was not significantly delayed. The NRC also concluded that the issue did not affect plant operation.

Both the Davis-Besse shield building and the Crystal River Unit 3 containment laminar cracking examples discussed above were identified while cutting construction openings for major equipment replacement. However, the NRC concluded that there were no other similarities between these two issues.

Alkali-Silica Reaction Concrete Degradation at Seabrook Station

Alkali-silica reaction is a slow chemical process that can occur over time in hardened concrete. For this reaction to occur, it is necessary for the concrete to contain reactive aggregate, high alkali content in the cement, and adequate moisture to form a gel. The gel expands by absorbing water initially, resulting in a network of micro-cracks in the concrete. Depending on its progression and severity, the alkali-silica reaction can reduce or affect mechanical properties of concrete (i.e., compressive, tensile, shear, and bond strengths, elastic modulus, and the Poisson's ratio) used in design to different extents, and could also affect empirical code relationships between mechanical properties in the American Concrete Institute design and construction codes. Alkali-silica reaction can potentially affect structural performance over time.

In August 2010, during an assessment for the license renewal application by the Seabrook Station, the licensee identified concrete degradation due to alkali-silica reaction in below-grade walls of several safety-related structures with ground water intrusion. Seabrook is the first nuclear plant in the U.S. nuclear power industry to identify this type of degradation. The licensee's root cause analysis determined that, along with other causal factors, the alkali-silica reaction developed in Seabrook's concrete primarily because the concrete mix had a susceptible aggregate that was slow-reacting. The potential reactivity of this aggregate was undetected by the testing specified by the applicable American Society for Testing and Materials construction standards at the time of construction in the late 1970s. Since that time, the role of slow-reacting aggregate in alkali-silica reaction has been identified in the construction industry and improved standard tests are now available to better identify slow reactive aggregates before use.

Seabrook has continued detailed testing, walkdowns, crack monitoring, and evaluations to comprehensively address and manage the issue in the short- and long-term. On May 16, 2012, the NRC staff issued a letter to the licensee to confirm commitments to comprehensively address this issue. The letter focuses on assuring operability of the affected structures pending review of a formal root cause analysis and short- and long-term monitoring actions while plant-specific alkali-silica reaction research and development continues. The research and development results will be used to address long-term effects on structural performance and management of the issue, and to provide the technical basis for the final operability determination and corrective actions, if required.

The NRC staff reviewing Seabrook's license renewal application is focusing on the discovery of this concrete degradation because the aging effects of alkali-silica reaction on the affected structures may be different in character or magnitude after the term of the current operating license. The licensee needs to demonstrate that the aging effects during the period of extended operation will be adequately managed. The NRC is currently evaluating the proposed plant-specific, first-of-a-kind, alkali-silica reaction aging management program.

The NRC staff's plant oversight reviews are focused on ensuring that the alkali-silica reaction issue at Seabrook is comprehensively addressed and managed such that there is reasonable assurance that the affected structures will continue to perform their intended safety functions through the expected service life. The staff has performed detailed inspections to verify and assess the adequacy of the licensee's interim operability basis and actions and commitments to address the impact on reinforced concrete structures at Seabrook. The NRC, through followup inspections, will verify the adequacy of actions related to the alkali-silica reaction structures monitoring program, and the proposed large-scale testing to reconcile this issue with the design and licensing basis.

The NRC has also engaged external stakeholders and members of the public through public meetings and written communications under the reactor oversight and license renewal processes. On November 18, 2011, the NRC issued IN 2011-20, "Concrete Degradation by Alkali-Silica Reaction," to inform licensees of the occurrence of alkali-silica reaction-induced concrete degradation of safety-related structures at Seabrook.
The NRC's oversight review of this issue determined that there are no immediate safety concerns based on existing safety margins, the localized and slow nature of the degradation, and ongoing crack monitoring. This review has included an evaluation of the licensee's prompt operability determinations for various structures affected by this reaction. The results of the NRC staff's review are documented in "Seabrook Station, Unit No. 1 - Confirmatory Action Letter Follow-Up Inspection - NRC Inspection Report 05000443/2012010," dated August 9, 2013.

Cumulative Effects of Regulation

The NRC began addressing ways to mitigate the cumulative effects of regulation in 2009, in response to Commission direction stemming from a meeting on the emergency preparedness final rule. Since then, the staff prepared two Commission papers: SECY-11-0032, "Consideration of the Cumulative Effects of Regulation in the Rulemaking Process," dated March 2, 2011 and SECY-12-0137, "Implementation of the Cumulative Effects of Regulation Process Changes," dated October 5, 2012. In response to SECY-12-0137, the Commission directed (among other items):

- Any expansion of the consideration of the cumulative effects of regulation should be considered in the broader context of actions directed from COMGEA-12-0001/COMWDM-12-0002, "Proposed Initiative to Improve Nuclear Safety and Regulatory Efficiency."

- The staff should continue to develop and implement outreach tools that will allow NRC to consider more completely the overall impacts of multiple rules, orders, generic communications, advisories, and other regulatory actions on licensees and their ability to focus effectively on items of greatest safety import.

- The staff should engage industry to seek volunteer facilities to perform "case studies" to review the accuracy of cost and schedule estimates used in NRC's regulatory analysis.

The staff is currently addressing the Commission direction, and will prepare a follow-on Commission paper by March 2015 that describes the effectiveness of the cumulative effect of regulation program. In addition, the NRC is applying the principles of the process enhancements, to the extent practicable, to the post-Fukushima regulatory actions. For instance, the NRC has engaged in significant public interaction (through public meetings, comment periods, etc.) during their development, and is providing implementation guidance when necessary.

Evaluation of Economic Consequences

The NRC's regulatory framework affords flexibility in accounting for the offsite economic consequences associated with unintended releases of radionuclides with subsequent land contamination. Specifically, consideration of offsite property damage can arise during cost-justified substantial safety enhancements (i.e., backfit analysis), as well as regulatory and environmental analyses. The NRC uses similar guidance documents to conduct the cost-benefit determinations of these analyses, and the staff is currently updating specific aspects of the guidance to reflect up-to-date data, such as NRC's current dollar per person-rem conversion factor policy and current replacement energy costs.

Counterfeit, Fraudulent, and Suspect Items

The integrity of the supply chain is a fundamental element of an effective quality assurance program for the NRC's licensed facilities and the suppliers of basic components to these facilities. During the late 1980s and early 1990s, the NRC and the commercial nuclear power industry performed a major reassessment of the supply chain in response to numerous occurrences in which counterfeit or fraudulent materials and components had entered the supply chain of NRC licensed facilities. The NRC issued several generic communications during that time to inform licensees and suppliers of these threats, and methods to identify counterfeit, fraudulent, and suspect items (CFSI), including steps to mitigate risk to the nuclear supply chain. Two key CFSI documents issued during that time period were:

- GL 1989-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products," dated March 21, 1989

- GL 1991-05, "Licensee Commercial-Grade Procurement and Dedication Programs," dated April 9, 1991.

These documents have remained effective for more than 2 decades, with little to no significant counterfeit activity evidenced in the U.S commercial nuclear industry to date.

However, other U.S. industries have seen an increase in CFSI activity in recent years. In 2010, the U.S. Department of Commerce published a study of the electronics supply chain supporting the U.S. Department of Defense. The report indicated that the electronics industry may be experiencing a far greater challenge today than the nuclear industry experienced in the 1990s. The report was based on an extensive survey of 387 original equipment manufacturers, original component manufacturers, electronics distributers, brokers, and suppliers to the U.S. Department of Defense. The survey was extensive, asking more than 80 procurement and quality related questions to assess the depth and breadth that counterfeiting had permeated the U.S. Department of Defense electronics supply chain. The survey showed a significant trend of a 120 percent rise in electronic counterfeiting since 2005. This trend appears to repeat itself in other heavily industrialized business sectors as well, including the petroleum, automotive, transportation, commercial airline, and the construction industries.

Several factors can influence the introduction of CFSI. Historically, obsolete parts have served as targets for CFSI. The buyers of rare or hard-to-find items have been known to pay large sums of money or assume unconventional levels of risk to prevent a process disruption at a plant or of a critical mission. However, the U.S. Department of Commerce study shifted that paradigm by reporting that obsolescence was only a factor in less than half of the reported counterfeit instances. The majority of recently documented cases were related to new items, commonly referred to as “in process” items. Counterfeiters have significantly upgraded their capabilities and skills to manufacture CFSI that are increasingly more difficult to detect.

A concern that factored into the NRC’s decision to evaluate the extent of CFSI was the industry’s transition from analog to digital instrumentation and controls technology. Along with the shift to more advanced technologies come the risks and vulnerabilities other industrialized business sectors are experiencing.

Based on interactions with the U.S. Nuclear Procurement Issues Committee and EPRI, the staff determined that the following factors were influencing CFSI:

- part standardization, making a product’s design vulnerable
- long, complex supply chains and a shift to a more globalized supplier base
- the advent of the Internet and increased use of alternate sourcing techniques
- internal quality assurance programs not focused on CFSI
- a sense of complacency based on the belief that someone else along the supply chain had been checking for CFSI
- using commercially manufactured parts or components in applications requiring high degrees of quality assurance

To address this growing concern, the NRC staff issued SECY-11-0154, “An Agencywide Approach to Counterfeit, Fraudulent, and Suspect Items,” dated October 28, 2011. The document identified several strategies and actions for the agency to take to further address CFSI. The Office of New Reactors has the lead to coordinate with other NRC offices and developed a formal agencywide strategy to monitor and evaluate CFSI. This strategy included the creation of a CFSI working group of representatives from potentially affected NRC offices. The working group identified 19 actions to respond to challenges associated with CFSI. These actions were categorized in the following areas to assist with the project planning efforts:

- supply chain oversight
- communications (both internal and external)
- agency response protocols
- cyber security supply chain oversight

The CFSI working group gathered and assessed information related to current counterfeiting activity, security risks and events, current practices in non-NRC-regulated activities, and proposed activities in NRC-regulated activities. The working group assessed operating experiences internal to the commercial nuclear industry, including information collected by the NEI and EPRI. External experience, such as that collected by the U.S. Department of Energy, the U.S. Department of Defense, the National Aeronautics and Space Administration, and the Aircraft Industry Association also were considered.

In response to SECY-11-0154, NEI has taken the lead to develop several industry voluntary initiatives to address key aspects in the prevention and control of CFSI in the nuclear supply chain. These industry initiatives are expected to encourage the use of standard procurement language in vendor or supplier purchase orders and service contracts for incorporating anti-CFSI processes and protocols, as well as reporting and sharing CFSI events. Industry is also developing guidance for effectively quarantining suspect parts to support followup investigations.

The NRC recently issued IN 2012-22, “Counterfeit, Fraudulent, Suspect Item (CFSI) Training Offerings,” dated January 25, 2013. The objective of this IN is to inform addressees of a sampling of the entities that offer training on how to detect potential CFSI that may enter the supply chain and to heighten awareness of CFSI issues.

In 2014, the staff intends to report on the status of addressing the 19 actions that the CFSI working group identified in Commission Paper, SECY-11-0154.
During 2012-13, several international incidents became evident regarding the existence of fraudulent certification documents used to authorize the use of commercially manufactured parts and components in safety related applications at commercial nuclear power plants. While similar instances were not identified in any of the U.S. plants, the topic was presented at the February 2013 meeting of the NRC’s Agencywide CFSI Steering Committee to make committee members aware of the fraudulent activity. This CFSI Steering Committee is comprised of Senior Executives from each of the NRC Offices potentially affected by CFSI.

Construction Inspection Program Lessons Learned

After issuing a combined license for a new reactor, in accordance with 10 CFR Part 52, the NRC implements a construction inspection program during the period between licensing and initial operation. The NRC issues inspection reports documenting the results of construction inspections.

Governing documents and procedures are developed by the Office of New Reactors. The NRC’s Region II Office located in Atlanta, Georgia, has the primary responsibility for implementing the construction inspection program. Region II dispatches as many as five resident construction inspectors to a new reactor site during the preoperational phase of construction to oversee the day-to-day activities of the licensee and its contractors, and may supplement this inspection staff with additional personnel from Region II, other regional offices, and headquarters technical staff, as needed, to ensure that the as-built facility conforms to the conditions of the license. NRC resources are carefully managed to ensure that construction inspection activities do not in any way detract from the ongoing oversight of operating reactors.

The NRC conducts vendor inspections to ensure that products and services furnished to U.S. reactors meet established regulatory requirements for quality and other safety factors.

The NRC began to fully implement its construction inspection program with the issuance of the licenses for Vogtle Units 3 and 4 in February 2012. The program was expanded to include V.C. Summer Units 2 and 3 when their licenses were issued in March 2012.

The NRC evaluates inspection results to identify lessons learned that can be used as feedback to improve the construction inspection program, to focus future inspection activities, and to inform licensees of needed improvement areas. Over the course of the first year of the full implementation of the program, the following lessons learned were identified:

- Design and configuration control - licensees must align with designers, suppliers, and constructors to achieve effective design control, configuration and change management and comply with 10 CFR Part 52 when making changes to the certified design.

- Supplier oversight - licensees must effectively oversee all contractors, subcontractors, and vendors to ensure that they are meeting regulatory and inspections, tests, analyses, and acceptance criteria requirements.

- Digital instrumentation and control - licensees must focus on digital instrumentation and control systems to ensure compliance with licensing commitments and address design verification and validation issues early on.

- Licensee's ultimate responsibility – the licensee holds the ultimate responsibility to meet its obligations under its license and must demonstrate that it is a competent and capable operator. Engineering, procurement, and construction contracts and their implementation must preserve these principles.

1.3.3 Major Regulatory Accomplishments

Since its previous U.S. National Report was issued in 2010, the NRC has been actively working on lessons learned from Fukushima and has issued four combined licenses. The NRC also amended its regulations concerning fatigue management and had major accomplishments in the areas of PRA, fire protection, and plant restart.

Fukushima Lessons Learned

As a result of the Japan event, the NRC has interacted with the U.S. nuclear industry, members of the public, and other stakeholders on how to best implement the actions contained in the NTTF report and additional Commission direction. The NRC's public Web site has a dedicated portal (http://www.nrc.gov/reactors/operating/ops-experience/japan-info.html) that contains information on activities and documents related to the agency's implementation of lessons learned from Fukushima.

In 2012, the NRC conducted more than 80 public meetings related to the lessons learned from the Fukushima accident. Numerous NRC representatives have presented information on Fukushima and the lessons learned to university scholars and professional scientific and engineering societies. The 2012 and 2013 NRC Regulatory Information Conferences also included presentations on activities associated with Fukushima lessons learned. These meetings, interactions, and exchanges of information have been instrumental in obtaining early input, which was factored into the NRC's regulatory activities and lessons learned implementation plans. This is a testament of the agency's commitment to maintain open and transparent regulatory activities.

On March 12, 2012, the NRC issued orders to nuclear power plant licensees for:

- developing and implementing mitigation strategies to respond to external events resulting in the loss of power at plants
- ensuring reliable containment vents at boiling-water reactors (BWRs) with Mark I and Mark II containments.
- enhancing SFP instrumentation

Also in March 2012, the NRC issued licensees a request for information to reevaluate seismic and flooding hazards, perform inspections (or “walkdowns”) of existing seismic and flooding protection features, and assess emergency communication systems and staffing levels.

In June 2013, the NRC modified the Order that required reliable hardened vents for BWRs with Mark I and Mark II containments to require that those vents remain functional in the conditions following reactor core damage. As directed by the Commission in its SRM related to SECY-12-0157, “Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments,” the NRC staff is also assessing possible changes to regulations through the rulemaking process to include strategies for filtering or otherwise confining radioactive material that gets released as a reactor core is damaged. These strategies would be in addition to the protections already in place. The rulemaking technical bases will include an evaluation of a variety of performance criteria, such as a decontamination factor, equipment and procedure availability similar to those required to implement 10 CFR 50.54(hh), or other measures that may be developed during stakeholder engagement. The performance and risks of the various filtering strategies and equipment considered will be fully evaluated, including an assessment of the benefits and costs associated with various approaches to severe accident management and filtering strategies. The regulatory analysis for the possible rulemaking assumes implementation of orders associated with improvements to mitigation strategies for beyond-design-basis events and the severe accident capable containment vents. If pursued, the rulemaking related to Mark I and Mark II containments would be developed and issued for public comment. An evaluation of all containment types will build upon the insights and analyses associated with the Mark I and Mark II designs.

Issuance of Combined Licenses

In 2012, the NRC issued two combined licenses – to Southern Nuclear Operating Company for Vogtle Units 3 and 4 in Georgia, and to South Carolina Electric and Gas for the V.C. Summer Units 2 and 3 in South Carolina. These are the first four combined licenses issued by the NRC and the first time this portion of the 10 CFR Part 52 licensing process has been implemented. According to this process, the NRC authorized the licensees to construct and (with specified conditions) operate the nuclear power plants. The combined licenses are valid for 40 years from the date of the Commission finding that the acceptance criteria in the combined license are met.

Managing Fatigue Rule Implementation

On March 31, 2008, the NRC published a rule that included new regulations in 10 CFR Part 26, Subpart I, “Managing Fatigue.” The NRC required licensees to implement the requirements in the rule by October 1, 2009, giving them an 18-month period to hire and train individuals as needed to ensure proper implementation of the fatigue management requirements. Subpart I strengthens the effectiveness of fitness for duty programs by requiring reasonable assurance that worker fatigue does not adversely affect public health and safety. It establishes enforceable requirements for the management of worker fatigue, including requirements for a minimum number of days off to ensure that individuals have opportunities for recovery sleep. In addition to the rulemaking and its associated analyses, the NRC issued RG 5.73, “Fatigue Management for Nuclear Power Plant Personnel,” in March 2009, to provide guidance on how to implement the rule.

Following implementation of the rule, the NRC received several petitions for rulemaking to modify 10 CFR Part 26, Subpart I. The petitioners alleged that implementation of the rule had impeded some beneficial safety practices. During a meeting to better understand the issues raised in the petitions, industry representatives stated that implementation of the minimum days off requirements achieved its goal of ensuring that workers are provided with adequate rest periods. They further noted that there have been undesirable consequences, in particular the industry's inability to continue practices that licensees consider beneficial, such as promoting consistency among work crews and the continued development of licensee staff. The industry further stated that the hours available for work are sufficient in almost all cases; however, they believe there should be more flexibility in how the time can be used. As a result of the meeting, the NRC gained an understanding of why the industry viewed the rule as complex and lacking flexibility.

The NRC worked with industry and other external stakeholders to develop an alternative method for managing cumulative fatigue that provides greater scheduling flexibility to licensees. The primary benefit of the alternative method is that is does not impede implementation of the beneficial safety practices that the industry has stated have been curtailed under the existing minimum days off requirements. The alternative method limits work hours to a weekly average of 54 hours worked, with work hours being averaged over a rolling period of up to 6 weeks. As a result, the alternative limits work hours to levels comparable to the original minimum days off requirements while adding the simplicity and flexibility the industry desired. Similar to the current minimum days off requirements, this alternative, when implemented with the other aspects of Subpart I, helps prevent most instances of cumulative fatigue by limiting the number of extended work weeks and work days. In those cases where extended schedules are unavoidable, the alternative method will limit their duration as a way to mitigate cumulative fatigue.

The rule codifying the alternative method was published on July 21, 2011, and the rule became effective on August 22, 2011. To date, several licensees have adopted the alternative method and feedback indicates that it has allowed the beneficial safety practices to be reinstituted at those facilities that adopted that alternative.

Grow Your Own Probabilistic Risk Assessment Analyst Program

The increasing use of PRA in regulatory matters (as described in Article 10) and in response to recent events, such as the Fukushima earthquake and tsunami and the North Anna seismic event, has created a larger demand for PRA expertise in the regulatory body and by plant licensees and operators. The NRC has responded to the challenge of development and retention of new risk analysts by initiating an in-house recruitment and training program. The Grow Your Own PRA Analyst program is tailored to meet agency staffing and training needs by building and maintaining a pool of qualified Reliability and Risk Analysts to address current and future risk assessment regulatory requirements. The program is designed to take internal candidates with demonstrated strengths in nonrisk areas, such as regulation, nuclear power engineering, and operations, and provide them the requisite training in various topics in the PRA field.

The scope of the program is expected to span all agency offices that have PRA expertise. Currently, the program has placed candidates into the Office of Nuclear Reactor Regulation and the Office of New Reactors. Future candidates are expected to be placed into the Office of Nuclear Regulatory Research and the Office of Nuclear Materials Safety and Safeguards.

The Grow Your Own PRA Analyst program requires a 3-year training commitment from candidates. Upon satisfactory completion of the program and passing a qualification board, the participants are guaranteed a promotion with the title of Reliability and Risk Analyst. The training program combines elements of formal classroom training, independent study, and on-the-job training as members of their assigned branches. The extensive training includes about two dozen formal classes (over 1,000 hours), about two dozen independent study activities, and several specific on-the-job assignments. Senior level personnel with risk-related expertise mentor the candidates as they progress through the program. The mentors serve as coaches and are assigned according to their specific areas of expertise. This program is unique because it includes work rotations to other offices and divisions involved in risk activities and a learning project, which aims to give practical, hands-on experience in preparing a PRA analysis in a real-world setting. A panel of PRA experts will review the project results to determine whether the participants have demonstrated an understanding of the concepts of PRA by performing a satisfactory analysis.

NFPA-805 Improvements

Nuclear power plants use multiple layers of fire protection features to keep fires from damaging plant safety systems. Some of these features include administrative controls (such as hotwork) fire barriers, fire detection systems, fire suppression systems (such as sprinklers), and manual firefighting capability (such as fire brigades). If a required element of fire protection is not available, the licensee must take appropriate compensatory actions. The NRC regularly inspects licensees' means of achieving and maintaining the reactor's safe shutdown capability in the event of a fire.

Licensees have two alternative regulatory approaches to manage their fire risk – a deterministic approach or a risk-informed and performance-based approach. Deterministic fire protection requirements seek to establish safety margins through the postfire survival of a single train of systems needed to shut down the reactor. These requirements, based on a set of postulated fires, were developed before the staff or the industry had the benefit of PRAs for fires and other recent technical advances such as fire modeling. The NRC lists these requirements in 10 CFR 50.48(b) and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50.

The NRC modified its regulations to allow licensees to adopt, on a voluntary basis, National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," in place of their existing fire protection licensing basis. Risk-informed fire protection requirements consider risk insights as well as other factors to establish requirements that better focus attention on design and operational issues according to their importance to public health and safety.

Performance-based regulations rely on a required outcome rather than requiring a specific process or technique to achieve the outcome. The NRC lists these requirements in 10 CFR 50.48(c). The NRC published the risk-informed and performance-based alternative regulation, 10 CFR 50.48(c) in June 2004, allowing licensees to focus their fire protection activities on the areas of greatest risk. The rule permits licensees to use the fire protection requirements contained in NFPA 805, with some exceptions. To help licensees transition from their current fire protection program to one based on NFPA 805, NRC staff issued RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 0, in May 2006. This NRC guide endorses the related NEI 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c),” Revision 1, issued in September 2005, because it provides the detailed methods acceptable to the NRC for implementing NFPA 805 and complies with regulations and regulatory positions outlined in the RG. The staff issued RG 1.205, Revision 1, in December 2009.

Implementing a risk-informed, performance-based fire protection program in accordance with NFPA 805 can lead to substantial improvements in reactor safety as a result of risk-significant modifications. NFPA 805 will also reduce the need for future exemptions and unnecessary regulatory burden associated with the current fire protection approaches and will maintain reactor safety while adding appropriate flexibility to licensees’ fire protection activities.

Regulatory Review to Support Plant Restart Following External Events

The NRC staff evaluates each external event individually to determine what level of review will be conducted to ensure public health and safety. The NRC maintains awareness of predicted or ongoing external events through a variety of mechanisms. These include: resident inspector and regional oversight, headquarters operational assessment personnel who monitor events daily, and notifications from licensees to the NRC Headquarters Operations Center. When there is a major external event that causes a plant shutdown to occur, the NRC staff will apply the appropriate process and regulatory requirements depending on the nature of the event. Specifically, NRC staff evaluates events to determine if a regulation or license condition governs restart, or if plant safety considerations necessitate regulatory action such as an order or confirmatory action letter to establish NRC control over the plant restart process. It should be noted that not all major external events require NRC approval for plant restart. However, for those that do, the staff will implement the appropriate regulatory framework to conduct its review, which will result in a thorough evaluation of plant conditions and safety before restart. In addressing these types of external events, NRC staff will implement guidelines in the Office of Nuclear Reactor Regulation Office Instruction LIC-504, “Integrated Risk-Informed Decision Making Process for Emergent Issues,” dated April 12, 2010. LIC-504 includes guidance on making and documenting risk-informed decisions regarding the action that the NRC should take in response to a potentially significant, emergent issue at a U.S. nuclear power plant. In addition, the NRC staff will implement guidelines in Office Instruction LIC-502, “Procedure for Development, Implementation, and Management of Action Plans,” dated September 30, 2009. LIC-502 provides guidance on when an action plan should be developed for evaluating and resolving an issue, including the need to identify appropriate actions, assessments, and milestones to adequately address any safety issues associated with the external event.

NRC Management Directive (MD) 8.3, “NRC Incident Investigation Program,” revised on March 27, 2001, includes guidance for determining the agency’s appropriate event response and delineates responsibilities at the office-level for response to significant operational events.

Depending on the circumstances surrounding the external event and the plant response, the NRC will tailor its response according to applicable regulations, management directives and Inspection Manual chapters. The agency’s actions are dictated by its mission of protecting the health and safety of the public and environment, and guided by our principles of openness and transparency.

Examples of how the NRC evaluates plant restart following external events can be found in two recent events: the restart of the North Anna Power Station after the August 2011, earthquake, and the current restart evaluation of the Fort Calhoun Station. On August 23, 2011, North Anna experienced ground motion from a magnitude 5.8 earthquake approximately 10-miles from the site. Subsequent analysis indicated that the spectral and peak ground accelerations for the operating basis and design-basis earthquakes were exceeded at certain frequencies for a short period of time. In accordance with Section V(a)(2) of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, a nuclear power plant is required to be shut down when the vibratory ground motion exceeds that of the operating basis earthquake. In addition, the regulations state that "prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public." To further support these requirements, the NRC issued a confirmatory action letter to the licensee confirming the licensee's commitment that North Anna Units 1 and 2 would not be restarted until the NRC had completed its review and authorized operation. To demonstrate that no functional damage occurred as a result of the earthquake and that it was safe to operate the facility without undue risk to the health and safety of the public, the licensee performed a number of inspections, tests, and analyses, consistent with EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," issued in December 1989, which the NRC had previously endorsed in RG 1.167, "Restart of a Nuclear Plant Shut Down by a Seismic Event," issued in March 1997, with certain exceptions. The NRC reviewed the licensee's summary report of the plant's response and concluded that the licensee acceptably demonstrated that no functional damage occurred at North Anna to those features necessary for operation, and that North Anna Units 1 and 2 could be operated without undue risk to the health and safety of the public.

Fort Calhoun Station initially shut down for a scheduled outage in April 2011. The shutdown was extended because of the summer 2011 flooding of the Missouri River and to address longstanding technical issues. As a result, the NRC increased its regulatory oversight of Fort Calhoun under the process for Inspection Manual Chapter 0350, "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns," dated December 15, 2006. Fort Calhoun is currently receiving increased NRC oversight and is required to obtain NRC approval for restart of the facility. The NRC issued a revised confirmatory action letter on February 26, 2013, documenting the actions that the plant will need to take prior to restart to address the issues that resulted in prolonged performance decline. Included in the confirmatory action letter was the Restart Checklist, which listed the specific items that the NRC will review. On March 7, 2013, the NRC issued a revised "U.S. Nuclear Regulatory Commission Manual Chapter 0350 Panel Fort Calhoun Station Restart Checklist Basis Document." This document provides details and clarification for the scope and breadth of the items included in the Restart Checklist and the actions that the NRC plans to take to verify that Fort Calhoun has adequately addressed them.

An external event resulting in significant damage to the offsite infrastructure in the vicinity of a nuclear power plant may degrade the capabilities of offsite response organizations in the 10-mile plume exposure planning zone. If an external event causes damage to offsite infrastructure to the extent that the continued adequacy of offsite emergency preparedness is seriously questioned, FEMA may perform a disaster initiated review to reaffirm the radiological emergency preparedness capabilities of affected offsite jurisdictions. This review is not intended to be a comprehensive review of offsite plans and preparedness, but rather to inform NRC decisions about plant restart, as discussed above. The interagency protocol for conducting this review is outlined in Section I of the Memorandum of Understanding between the NRC and FEMA, contained in Appendix A, "Memorandum of Understanding between Federal Emergency

Management Agency and Nuclear Regulatory Commission," to 44 CFR Part 353, "Recovery from Disasters Affecting Offsite Emergency Preparedness."

Emergency Preparedness Rule Implementation

On November 23, 2011, the NRC amended its emergency preparedness regulations; the amendments to the regulations became effective on December 23, 2011. The rulemaking amended the emergency preparedness regulations in 10 CFR Part 50 and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. The rulemaking also amended other licensee emergency plan requirements. The new requirements include:

- enhancing the ability of licensees to prepare for and take certain emergency preparedness and protective measures in the event of a radiological emergency
- addressing, in part, security issues identified after the terrorist attacks of September 11, 2001.
- clarifying regulations to effect consistent emergency plan implementation among licensees
- modifying certain emergency preparedness requirements to be more effective and efficient

Specifically, the rulemaking included changes in the following areas:

- analysis of onshift staffing levels
- emergency action levels for hostile action events
- emergency response organization augmentation and alternate facilities
- licensee coordination with offsite response organizations during hostile action events
- protection for onsite personnel during hostile action events
- challenging drills and exercises to include hostile action events, no or minimal release scenarios, and rapidly progressing scenarios
- backup means for alert and notification systems
- emergency declaration timeliness
- emergency operations facility – performance-based approach
- evacuation time estimate updating
- amended emergency plan change process

The rulemaking process was open and transparent and included numerous opportunities for public and stakeholder participation. In conjunction with FEMA, the NRC engaged stakeholders in various ways during the development of this rule. The NRC and FEMA held numerous public meetings to discuss the proposed changes to the regulations. These meetings included participants from the nuclear industry, nongovernmental organizations, State and local agencies, members of the public, and other interested stakeholders. The NRC also requested formal public and stakeholder comments and considered these comments during the development of the new rule. The NRC and FEMA are updating several emergency preparedness guidance documents to reflect the changes in the NRC regulations.

1.4 International Peer Reviews and Missions

The United States strongly supports international peer reviews and the IAEA's suite of missions, including the CNS peer review activities, and the Integrated Regulatory Review Service (IRRS) and Operational Safety Review Team (OSART) missions. This section provides a summary of the results of the missions and peer review activities conducted since the last U.S. National Report was issued.

1.4.1 Convention on Nuclear Safety

The United States ratified the CNS in 1999 and has been actively participating in its peer review activities. The conclusions from the review of the 2010 U.S. National Report at the Fifth CNS review meeting in April 2011 were very positive.

Items Resulting from Country Group Session

A review of the questions raised by other contracting parties on the 2010 U.S. National Report identified the following areas of interest:

- staffing, training, and knowledge management
- new reactor licensing
- safety culture
- human factors
- digital instrumentation and control
- license renewal and aging management
- Reactor Oversight Process
- emergency preparedness
- audits and vendor inspections
- quality assurance
- risk-informed regulations
- PRA
- IRRS
- international standards
- radiation protection
- financial considerations

The NRC's presentation during the 2011 review meeting focused on these topics. The Institute of Nuclear Power Operations (INPO) also discussed its role in maintaining and improving nuclear safety. Both the NRC and INPO discussed actions taken immediately after the Fukushima accident.

The United States was a member of Country Group 1. The group participants concluded that the United States implemented the following good practices:

- an effective approach to ensure safety by combining the experience from NRC, a mature regulatory body, with the strong self evaluation processes provided by INPO
- an enhanced knowledge management system and hiring policies to effectively incorporate a new generation of experts
- formalized policies and procedures for maintaining openness and transparency in all regulatory activities, including hosting the Regulatory Information Conference, which allows effective public and stakeholder engagement

Country Group 1 identified the following challenges for the United States:

- continue addressing buried piping degradation and ground water protection issues
- continue evaluating and responding to cyber security threats

- enhance the digital instrumentation and control licensing process
- enhance the safety and security interface

Country Group 1 highlighted the following planned United States initiatives:

- finalize the action plan in response to IRRS finding
- address buried piping and ground water protection issues
- finalize the emergency preparedness rulemaking

The current U.S. National Report addresses these issues under the relevant articles.

1.4.2 Integrated Regulatory Review Service

The NRC regularly provides technical experts to participate in IRRS missions around the world, often at a senior leadership level. The NRC also hosted an IRRS mission in October 2010. The mission report contains 2 recommendations, 20 suggestions, and 25 good practices. The NRC will be hosting the followup mission in early 2014, as discussed in greater detail in Section 8.1.5.2.

1.4.3 Operational Safety Review Team

The NRC regularly provides technical experts to participate in OSART missions around the world, often at a senior leadership level. In June 2011, Seabrook Station Unit 1 hosted an OSART mission. The OSART team found a number of areas of good performance including a healthy reporting culture. A followup OSART mission was hosted in June 2013, as discussed in greater detail in Section 8.1.5.3.

PART 2

ARTICLE 6. EXISTING NUCLEAR INSTALLATIONS

Each Contracting Party shall take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention enters into force for that Contracting Party is reviewed as soon as possible. When necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible. The timing of the shutdown may take into account the whole energy context and possible alternatives, as well as the social, environmental, and economic impact.

This section explains how the United States ensures the safety of nuclear installations in accordance with the obligations in Article 6. It covers the reactor licensing and major oversight processes in the United States. This section also discusses programs for rulemaking, fire protection regulation, decommissioning, research, programs for public participation, and lessons learned from Fukushima. The U.S. Nuclear Regulatory Commission (hereafter referred to as the NRC, Commission, agency, or staff) posts the major results of assessments on the agency's public Web site at http://www.nrc.gov.

6.1 Introduction

The mission of the NRC is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to protect public health and safety, promote the common defense and security, and protect the environment. The NRC's primary goal is safety. The agency achieves this goal by ensuring that licensee performance is at or above acceptable safety levels. The NRC's licensees are responsible for designing, constructing, and operating nuclear facilities safely, while the NRC is responsible for the regulatory oversight of the licensees. Five strategic outcomes for this goal are specified:

(1) No nuclear reactor accidents.
(2) No inadvertent criticality events.
(3) No acute radiation exposures resulting in fatalities.
(4) No releases of radioactive materials that result in significant radiation exposures.
(5) No releases of radioactive materials that cause significant adverse environmental impacts.

The NRC met all of its safety strategic outcomes in fiscal years (FYs) 2010, 2011 and 2012.

The NRC also uses performance measures to determine whether the agency has met its safety goal. The NRC met its performance measures in FYs 2010, 2011, and 2012. Currently the NRC uses six performance measures.

The first measure analyzes nuclear power plant performance based on a large number of performance indicators and inspection findings.

The second measure tracks significant precursor events at nuclear power plants determined by the likelihood of an event adversely impacting safety.

The third performance measure indicates whether the NRC identifies significant issues in a nuclear power plant during inspections conducted under the Reactor Oversight Process.

The fourth measure tracks the trends of several key indicators of nuclear power plant safety. This measure is the broadest measure of the safety of nuclear power plants, incorporating the performance results from all plants to determine industry average results.

These four measures indicated that the nuclear power plants were safely operated, and that the events that did occur were of relatively minor significance.

The other two measures address harmful radiation exposures to the public and occupational workers and radiation exposures that harm the environment. Neither of these measures exceeded their targets in FYs 2010, 2011, and 2012.

6.2 Nuclear Installations in the United States

Annex 1 to Part 2 of this report lists all 104 nuclear installations in the United States, as discussed in NUREG-1350, Volume 24, "Information Digest 2012-2013," issued August 2012, which is available on the agency's Web site.

Appendix A of NUREG-1350 also lists installations in the United States that are under active construction or deferred plant status per the Commission's Policy Statement on Deferred Plants. Bellefonte Nuclear Plant, Units 1 and 2, are currently in deferred status. Watts Bar Nuclear Plant, Unit 2, is under active construction and the staff is reviewing its operating license application. For additional information please refer to Section 18.1.2.

The combined licenses for Vogtle Units 3 and 4 were issued in February 2012. Combined licenses for V.C. Summer Units 2 and 3 were issued in March of 2012. The four Westinghouse AP1000 reactors are currently under construction. The NRC is overseeing their construction using its construction inspection program for units licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Additional information on Vogtle and V.C. Summer construction activities can be found in Articles 18 and 19.

On February 5, 2013, following a comprehensive analysis, Duke Energy announced that it would permanently cease operation of Crystal River Unit 3. By letter dated February 20, 2013, Florida Power Corporation, the licensee for Crystal River Unit 3 and a subsidiary of Duke Energy, submitted a letter in accordance with 10 CFR 50.82, "Termination of License," certifying that Crystal River Unit 3 would cease power operations and that all fuel had been permanently removed from the reactor. Upon docketing of this letter, the 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," license for Crystal River Unit 3 no longer authorizes operation of the reactor or emplacement or retention of the fuel in the reactor vessel. In a press release issued on February 5, 2013, Duke Energy stated that they intend to use the SAFSTOR option for decommissioning. Generally, this involves placing the facility into a safe storage configuration, requiring limited staffing to monitor plant conditions, until the eventual dismantling and decontamination activities occur, usually in 40 to 60 years.

On February 25, 2013, Dominion Energy Kewaunee, the licensee for Kewaunee Power Station, certified May 7, 2013, as the date for permanent cessation of power operations at Kewaunee. In its press release of October 22, 2012, Dominion Energy cited economic reasons for closure, including the low cost of natural gas. Dominion Energy has selected SAFSTOR as its method of decommissioning and 60 years after permanent cessation as the duration within which to complete decommissioning. On May 7, 2013, Kewaunee Power Station permanently ceased operations, and on May 14, 2013, the plant completed and certified permanent defueling of the reactor vessel. Dominion Energy is no longer authorized operation of the reactor or to place fuel in the reactor vessel. Thus, on May 15, 2013, Kewaunee Power Station started the period of permanent cessation of operations and began the decommissioning process, which is planned, consistent with the regulations, to be completed within 60 years. On February 28, 2013, the NRC received Kewaunee's Post-Shutdown Decommissioning Activities Report. The report contained an overview of the activities and associated schedules planned for the plant's decommissioning and spent fuel management during the 60-year period. The licensee began major decommissioning activities 90 days after the NRC received the report. Major licensing activities currently include an amendment to modify Kewaunee's license and Technical Specifications and several exemption requests to adapt various 10 CFR regulations appropriate for a decommissioning facility. Licensing actions already submitted are expected to be completed by mid- to late-2014.

On June 12, 2013, Southern California Edison, the licensee for San Onofre Nuclear Generating Station Units 2 and 3, certified June 7, 2013, as the date for permanent cessation of power operations at San Onofre. In its press release, the licensee cited economic reasons for closure, including plant restart uncertainties brought by a recent ruling by the Atomic Safety and Licensing Board, which is an adjudicatory arm of the NRC. By letters dated June 28, and July 22, 2013, Southern California Edison submitted its certifications of permanent removal of fuel from the Unit 3 and Unit 2 reactor vessels, respectively. Southern California Edison is no longer authorized to operate the San Onofre reactors or to place fuel in the reactor vessels. Southern California Edison has not yet notified the NRC which decommissioning option will be selected. The licensee is not allowed to commence major decommissioning activities until 90 days after submitting the required Post-Shutdown Decommissioning Activities Report, which may be submitted up to two years after the permanent cessation of operation.

On August 27, 2013, Entergy Corporation announced that it plans to close and decommission its Vermont Yankee Nuclear Power Station. The station is expected to cease power production after its current fuel cycle and move to safe shutdown in the fourth quarter of 2014. Entergy stated that making the decision now and operating through the fourth quarter of 2014 allows time to properly plan for a safe and orderly shutdown and prepare filings with the NRC regarding shutdown and decommissioning.

Entergy will establish a decommissioning planning organization responsible for planning and executing the safe and efficient decommissioning of the facility. Entergy has selected SAFSTOR as its method of decommissioning.

6.3 Regulatory Processes and Programs

6.3.1 Reactor Licensing

To construct and operate a new nuclear reactor, an entity must submit an application to the NRC for a license. If the NRC staff accepts the application, the staff will conduct a safety and environmental review. The public has opportunities to participate through a hearing process. The NRC licensed all current operating nuclear plants under the detailed two-step process, specified in 10 CFR Part 50, first issuing a construction permit and then an operating license. Since 1976, the NRC has not received applications to construct a new reactor under 10 CFR Part 50. A new single-step process, specified in 10 CFR Part 52, provides direction for issuing a combined license for construction and operation of a new reactor. The NRC has received 18 combined license applications for 28 reactors. To date, five of those combined license applications have been suspended at the request of the applicant and one combined license application was withdrawn. The NRC issued the first combined licenses in 2012, authorizing the construction and operation of four new units at two nuclear power plant sites in the United States. Regulations in 10 CFR Part 52 also provide for the issuance of design certifications that can be referenced in a combined license application. To date, the NRC has issued four design certifications and two design certification amendments. The industry has submitted applications for three additional design certifications and two design certification renewals. As specified in 10 CFR Part 52, the NRC can also issue an early site permit to approve a site for a domestic nuclear power plant independent of an application for a combined license. Early site permits are valid for 10 to 20 years and can be renewed for an additional 10 to 20 years. To date, the NRC has issued four early site permits and two limited work authorizations which allow the permit holder to perform limited construction activities at a site. Article 18 provides more detail about the 10 CFR Part 52 regulations.

The NRC's reactor licensing process provides for the review and approval of changes after initial licensing. The process allows amendments to the operating license to support plant changes, license renewal, changes of ownership and license transfer, exemptions and relief from NRC regulations, and increases in the reactor power level (i.e., power uprates). This report provides additional discussion of the process in the introduction and other articles (i.e., Articles 14, 17 and 18).

6.3.2 Reactor Oversight Process

Through its Reactor Oversight Process, the NRC continuously oversees nuclear power plants to verify that they are being operated safely and in accordance with the agency's rules and regulations. The NRC has full authority to take actions necessary to protect public health and safety and may demand immediate licensee actions, up to and including a plant shutdown, to address declining or unacceptable safety performance at a domestic nuclear power plant.

The Reactor Oversight Process monitors plant performance in three key areas: reactor safety, radiation safety, and safeguards. Within these three areas are seven cornerstones of safety and security: initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and security. The Reactor Oversight Process assesses plant performance across the seven cornerstones using both inspection findings and performance indicators. At least two resident inspectors are stationed at each plant to perform routine inspections and provide immediate response to events. Additional inspectors from the NRC's regional offices perform more specialized inspections in areas like fire protection, operator licensing, security, and other aspects of plant design and operation. The NRC's baseline inspection program is the minimum regimen of inspection that is conducted at every U.S. nuclear plant. The inspection program is augmented by performance indicators to provide insights into licensee performance. Under the performance indicator program, each licensee compiles performance data and provides it to the NRC quarterly. The NRC posts plant-specific inspection findings and performance indicator information on the agency's public Web site.

The NRC uses an Action Matrix to objectively determine its regulatory response to plant performance based on the inspection findings and performance indicators. The Action Matrix directs increased NRC oversight as licensee performance declines. If performance indicators exceed established thresholds or inspectors identify performance issues that involve more than nominal risk to public health and safety, the NRC may perform supplemental inspections and take additional actions to ensure that the licensee effectively resolves the degraded performance. The Action Matrix classifies plant performance in five columns, ranging from Column 1, which represents that the objectives of all cornerstones are met, to Column 5, which represents unacceptable performance. The NRC communicates its assessment of licensee performance on the public Web site, in publicly-available assessment letters to licensees, and in public meetings that are conducted annually. Individual plant performance information and additional information about the Reactor Oversight Process can be accessed through the following Web site: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html.

At the end of calendar year 2012, the nuclear reactors operating in the United States were distributed in the Action Matrix as follows:

- 99 reactor units were performing in the two highest performance categories of Reactor Oversight Process Action Matrix (Columns 1 and 2)
- 3 reactor units were at the third level of performance, the Degraded Cornerstone Column (Column 3),
- 1 reactor unit was in the Multiple/Repetitive Degraded Cornerstone Column (Column 4)

In addition, one reactor unit was in extended shutdown related to significant performance issues, and its performance was monitored outside of the Reactor Oversight Process (and associated Action Matrix) and under the oversight provisions of Inspection Manual Chapter 0350.

The NRC assesses the Reactor Oversight Process annually and evaluates its overall effectiveness in meeting pre-established program goals and intended outcomes. The NRC issued its latest report on the subject, SECY-13-0037, "Reactor Oversight Process Self-Assessment for Calendar Year 2012," on April 5, 2013. In this document the staff noted that:

- the performance indicator program continued to offer insights into ensuring plant safety and security, and the staff made several improvements to performance indicator program guidance and implementation in calendar year 2012
- NRC inspectors independently verified that licensees operated plants safely and securely, and the staff improved the inspection program through ongoing enhancements to inspection procedures and continual integration of operating experience into the inspection program
- the significance determination process continued to be an effective tool for determining the safety and security significance of inspection findings, and the staff made several improvements to governing guidance and significant progress on other significance determination process initiatives
- the assessment program ensured that the NRC and licensees took appropriate actions to address performance issues in calendar year 2012, commensurate with their safety significance

The self-assessment results for calendar year 2012 indicated that the Reactor Oversight Process continued to be an objective, risk-informed, understandable, and predictable process that was open to the public and effective in supporting the agency's mission and strategic goals of ensuring safety and security. In addition, the staff successfully reintegrated the Security Cornerstone into the assessment program as described in SECY-11-0073, "Staff Proposal to

Reintegrate Security into the Action Matrix of the Reactor Oversight Process Assessment Program."

The Reactor Oversight Process has developed into a mature oversight process over the past 12 years; however, the staff recognizes the value of continuous improvement and actively solicits internal and external stakeholder feedback to apply lessons learned and improve various key program areas of the Reactor Oversight Process.

6.3.3 Industry Trends Program

The NRC staff has implemented the Industry Trends Program since 2001, to confirm safe operation of the nuclear power plants at an industry level and to increase public confidence in the effectiveness of the NRC's processes. The agency uses industry-level indicators to identify adverse trends in performance. After assessing industry trends for safety significance, the NRC responds to any identified safety issues, including adjusting the inspection and licensing programs. Inspection Manual Chapter 0313, "Industry Trends Program," dated May 29, 2008, provides more detail about the program.

The Reactor Oversight Process uses both plant-level performance indicators and inspections to provide plant-specific oversight of safety performance, whereas the Industry Trends Program provides a way to assess overall industry performance using industry-level indicators. The NRC evaluates issues identified through either program using information from agency databases and addresses those determined to have generic safety significance, including generic safety inspections under the Reactor Oversight Process, the generic communications process, and the generic safety issue process.

One important output of the Industry Trends Program is the annual agency performance measures reported to Congress on the number of statistically significant adverse industry trends. The NRC Performance and Accountability Report includes this outcome measure. The latest report, NUREG-1542, Volume 18, "Performance and Accountability Report – Fiscal Year 2012," was issued in November 2012.

Based on the information currently available from the industry-level indicators and the Accident Sequence Precursor Program (discussed in Section 6.3.4 of this report), no statistically significant adverse industry trends were identified in FY 2012.

In addition to long-term trending of the data to identify statistically significant adverse trends, the NRC staff uses a statistical approach based on prediction limits to identify potential short-term, year-to-year emergent issues before they become long-term trends. None of the indicators exceeded its short-term prediction limits in FY 2012.

In 2008, the NRC staff implemented the Baseline Risk Index for Initiating Events (BRIIE) as part of the Industry Trends Program. The BRIIE (1) tracks several types of events that could potentially initiate a challenge to a plant's safety systems, (2) assigns a value to each initiating event according to its relative importance to the plant's overall risk of damage to the reactor core, and (3) calculates an overall indicator of industry safety performance.

The first level (referred to as Tier 1 performance monitoring) tracks and counts the number of times the initiating events that have an impact on plant safety occur in nuclear power plants during the year. The number of times that each event occurs is compared with a predetermined number of occurrences for that event. If the predetermined number is exceeded, one can infer possible degradation of industry safety performance. This annual tracking allows the NRC to intervene and engage the nuclear industry before any long-term adverse trends in performance emerge.

The second level (referred to as Tier 2 performance monitoring) addresses the risk to plant safety and core damage that each of the initiating events contributes. Each of the events is assigned an importance value, a ranking according to its relative contribution to overall risk to plant safety. The greater the contribution of the event to overall risk, the higher the importance value that is assigned to the event. Using statistical methods, the importance values are combined with the number of times the events occur during the year to calculate a number that indicates how much the overall industry risk of damage to the reactor core has changed from a baseline value. The NRC Performance and Accountability Report notes if this combined industry value reaches or exceeds a threshold value of 1×10^{-5} per reactor critical year, along with actions that have already been taken or are planned in response.

None of the initiating events tracked in Tier 1 exceeded its prediction limit in FY 2012. The BRIIE Tier 2 combined industry value in FY 2012 is below the established reporting threshold, and it indicates better than baseline industry performance.

SECY-13-0038, “FY 2012 Results of the Industry Trends Program for Operating Power Reactors,” dated April 8, 2013, which is available on the NRC public Web site, provides more details on the Industry Trends Program results for FY 2012.

6.3.4 Accident Sequence Precursor Program

The Accident Sequence Precursor Program systematically evaluates U.S. nuclear power plant operating experience to identify, document, and rank the operating events most likely to lead to inadequate core cooling and severe core damage (i.e., precursors).

To identify potential precursors, the NRC reviews plant events from licensee event reports and inspection reports. The staff then analyzes any identified potential precursors by calculating the probability of an event leading to a core damage state. A plant event can be one of two types, either (1) an occurrence of an initiating event, such as a reactor shutdown or a loss of offsite power, with or without any subsequent equipment unavailability or degradation, or (2) a degraded plant condition, depicted by the unavailability or degradation of equipment without the occurrence of an initiating event.

The Accident Sequence Precursor Program considers an event with a conditional core damage probability or an increase in core damage probability greater than or equal to 1×10^{-6} to be a precursor. The Accident Sequence Precursor Program defines a *significant* precursor as an event with a conditional core damage probability or an increase in core damage probability greater than or equal to 1×10^{-3}.

The Accident Sequence Precursor Program has the following objectives:

- Provide a comprehensive, risk-informed view of nuclear power plant operating experience and a measure for trending nuclear power plant core damage risk.

- Provide a partial check on dominant core damage scenarios predicted by probabilistic risk assessments (PRAs).

- Provide feedback to regulatory activities.

The NRC also uses the Accident Sequence Precursor Program to monitor performance against the safety goal established in the agency's Strategic Plan. Specifically, the program provides input to the following performance measures:

- Zero events per year identified as a significant precursor of a nuclear reactor accident.
- No more than one significant adverse trend in industry safety performance (determination principally made from the Industry Trends Program but supported by Accident Sequence Precursor results).

The staff completed precursor trend analyses as part of the annual Accident Sequence Precursor Program status report provided to the Commission in SECY-12-0133, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated October 4, 2012. The report provided insights such as the following:

- No significant precursors were identified in FY 2012. The last significant precursor was identified in FY 2002 (i.e., multiple degraded conditions at Davis-Besse).
- The staff detected no statistically significant trend for all precursors during the FY 2002–2011 period.

During the same period, the staff detected no statistically significant trends for precursor subgroups. These subgroups include precursors with a high safety significance (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-4}), initiating events, degraded conditions, loss of offsite power initiating events, precursors at boiling-water reactors (BWRs), and precursors at pressurized-water reactors (PWRs).

6.3.5 Operating Experience Program

The NRC recognizes that the effective use of operating experience is important for the agency's safety mission. Under the current NRC Strategic Plan, the agency is committed to "evaluate domestic and international operating events and trends for risk significance and generic applicability in order to improve NRC programs" as part of its effort to achieve the goal of safety. As a result, the NRC's emphasis on the effective use of operating experience remains strong.

The fundamental aim of the Operating Experience Program is to collect, evaluate, communicate, and apply operating experience information to achieve the NRC's principal safety mission of protecting people and the environment. Operating experience is reported to the NRC in licensee event notifications and in many other reports submitted under licensee reporting requirements, and described in reports of operating experience at foreign facilities. Sources of foreign operating experience include events submitted under the International Nuclear and Radiological Event Scale and reports submitted to the International Reporting System for Operating Experience. NRC staff systematically screens nuclear reactor-related operating experience for safety significance and generic implications. The NRC staff also determines the need for further action and application of lessons learned related to plant operating experience.

To support its safety mission, the NRC has resources dedicated to the review of operating experience. The NRC collects, stores, screens, and communicates operating experience;

conducts and coordinates the evaluation of operating experience; tracks the application of operating experience lessons learned; and coordinates NRC operating experience activities with other organizations performing related functions.

Upon launching the program, the NRC developed an internal Web site to provide a centralized source for accessing reactor operating experience information. This Web site is a gateway to the agency's operating experience document collections, contacts, search tools, sources, and reference material. In addition, the NRC created an operating experience community forum to quickly disseminate operating experience to the appropriate technical staff. The agency's public Web site contains all of the event reports licensees have submitted to the NRC.

Section 19.7 of this report provides more information about this program.

6.3.6 Generic Issues Program

The U.S. Congress mandated the NRC's agencywide Generic Issues Program to address issues that have significant generic implications related to safety or security that cannot be more appropriately addressed by other regulatory programs or processes. Sources of candidate generic issues include safety evaluations, operational events, and suggestions from NRC staff members, outside organizations, or members of the general public. Other existing programs generally address emergent issues that demand immediate attention (e.g., issues that may require plant shutdown) so that quick decisions can be made. The NRC maintains a complete list of all generic issues in NUREG-0933, "Resolution of Generic Safety Issues," published most recently in December 2011.

To efficiently use program resources and promote timeliness, the following seven criteria describe those issues that are appropriate for processing through the program:

(1) affects public health and safety, security, or the environment (this includes a risk threshold)
(2) applies to two or more facilities
(3) is not being addressed through other regulatory processes or voluntary industry initiatives
(4) can be resolved by new or revised regulation, policy, or guidance
(5) risk or safety significance can be adequately determined or estimated
(6) well defined and discrete
(7) may involve review, analysis, or action by the licensee

Substantial changes to the Generic Issues Program were instituted in 2007. Those changes were effective in clarifying roles and responsibilities, increasing participation in the process and establishing clear interfaces between the Generic Issues Program and other NRC processes. In 2012, the NRC applied business process improvement techniques to the program. Methods to resolve Generic Issues more efficiently and to improve communication on the progress of Generic Issues have been identified and are being implemented.

6.3.7 Rulemaking

The NRC's rulemaking process is used to impose new or to revise current requirements that licensees must meet to obtain or retain a license or certificate to use nuclear materials or to operate a nuclear facility. The NRC initiates a rulemaking in response to a congressional mandate, an Executive Order, a petition for rulemaking from outside the NRC, an internal recommendation from the technical staff, or direction received from the Commission in the form of a Staff Requirements Memorandum. Typically, the NRC publishes a proposed rule in the *Federal Register* for public comment. The public is usually given 75 to 90 days to provide written comments. Generally, all rules are issued for public comment. Those rules excepted from public comment deal with agency organization, procedure, or practice; are interpretive rules or general statements of policy; or are rules for which delaying their publication to receive comments would be contrary to public interest, unnecessary, or impracticable. Once the public comment period has closed, the staff analyzes the comments, makes any needed changes, and forwards the final rule for approval, signature, and publication in the *Federal Register*.

The NRC uses the Web site http://www.regulations.gov to provide an easy way for members of the public to access and comment on NRC rulemaking actions. The Web site contains proposed and final rulemakings that have been published in the *Federal Register* and any comments received, petitions for rulemaking, and other types of documents related to rulemaking proceedings.

The Commission must approve each final rule that involves matters of policy. The Executive Director for Operations is authorized to approve final rules that are minor, corrective, or nonpolicy in nature. Once approved, the final rule is published in the *Federal Register* and usually will become effective 30 days after the date of publication. Final rules that are considered major (e.g., have a significant impact on the economy) become effective at least 60 days after the date of publication. The section of this report on major regulatory accomplishments summarizes the significant nuclear reactor-related rules issued since the previous U.S. National Report.

6.3.8 Fire Protection Regulation Program

The NRC has three main foci in fire protection regulation: (1) implementation of the new risk-informed, performance-based fire protection licensing basis (10 CFR 50.48(c)); (2) resolution of the fire-induced multiple spurious operation and circuit analysis issue; and (3) resolution of licensees' nonconforming postfire operator manual actions. To support the implementation of 10 CFR 50.48(c), the NRC issued Regulatory Guide (RG) 1.205, and NUREG/CR-6850, Supplement 1, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," published in September 2010, reflecting lessons learned from the pilot application reviews. Two nuclear stations, Shearon Harris and Oconee, volunteered as pilot plants for the transition. The NRC reviewed these license amendment requests and issued safety evaluations in May and December 2010, respectively. By 2010, approximately one half of the U.S. reactor units had committed to transition to 10 CFR 50.48(c). The NRC also developed guidance to conduct triennial fire inspections of plants after they complete their transitions to the 10 CFR 50.48(c) licensing bases. As of August 2013, there are 19 license applications under review to implement 10 CFR 50.48(c) representing 29 units. Another 7 license applications representing 11 units are expected.

Nuclear power plants that are not transitioning to, or have not completed, their transitions to the risk-informed, performance-based fire protection rule are regulated under their current licensing bases. RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, issued October 2009, provides regulatory guidance for licensees on fire protection issues, including the treatment of fire-induced circuit failures and operator manual actions in response to fire damage. Plants that are not transitioning to, or have not completed their transition to, National Fire Protection Association (NFPA) 805 are inspected under the Inspection Procedure (IP) 71111.05T, "Fire Protection (Triennial)," dated April 21, 2006, and findings are evaluated using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005. Fire protection enforcement discretion has ended for sites not transitioning to 10 CFR 50.48(c).

The NRC's fire research program develops the technical bases for ongoing and future regulatory activities in fire protection and fire risk analysis. The NRC's current research program includes the following activities:

- developing and improving fire risk analysis methods and tools
- collecting, generating and analyzing fire-related data
- verifying, validating and improving fire models for regulatory use
- performing specialized fire testing on electrical cables for both hot shorts and fire properties
- evaluating shipping casks for beyond-design-basis fire conditions
- evaluating methods to predict operator performance during fire conditions
- providing specialized training on the fire PRA and for performing fire modeling

The fire research program supports the agency's strategic goals of safety and effectiveness and partners with other organizations such as the National Institute of Standards and Technology, the Electric Power Research Institute (EPRI), the University of Maryland, and international groups such as the Organisation for Economic Co-operation and Development Committee on the Safety of Nuclear Installations.

6.3.9 Decommissioning

The decommissioning process consists of a series of integrated activities, beginning with the nuclear facility transitioning from "operation" to "decommissioning" status and concluding with termination of the license, and release of the site. The NRC has adopted extensive regulations to ensure that decommissioning is accomplished safely and that residual radioactivity is reduced to a level that permits release of the property for either unrestricted or restricted use (Subpart E, "Radiological Criteria for License Termination," to 10 CFR Part 20). The NRC reviews and approves license termination plans, conducts inspections, processes license amendments, and monitors the status of decommissioning activities to ensure that radioactive contamination is reduced or stabilized. In addition, the decommissioning process includes several opportunities for public involvement.

In 2011, the NRC issued the Decommissioning Planning Rule, which updated 10 CFR 20.1406, "Minimization of Contamination," and 10 CFR 20.1501, "General." The design criteria for new facility construction discussed in 10 CFR 20.1406 requires applicants to describe how facility design and procedures will facilitate eventual decommissioning and minimize, to the extent practicable, the release of radioactive materials to the environment and the generation of radioactive waste.

To strengthen future decommissioning at existing operating facilities, 10 CFR 20.1501 requires surveys to identify contamination that would require remediation for license termination.

Guidance implementing the rule was provided in RG 4.22, "Decommissioning Planning during Operations," issued December 2012. The IAEA safety standards are a useful point of reference for future decommissioning provisions in the conceptual design of nuclear facilities.

NRC regulations and guidance (e.g., NUREG-1577, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1, issued February 1999) describe requirements and processes to review power reactor licensee financial qualifications and methods of providing decommissioning funding assurance. The regulations, as stated in 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," explain the requirements for decommissioning funding and decommissioning funding assurance.

Previously, the NRC determined that spent fuel can safely remain stored in the spent fuel pools (SFPs) or in dry cask storage facilities until a geologic repository is built and operating. The NRC regulations in 10 CFR Part 50 and 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, Reactor-Related Greater than Class C Waste," contain licensing requirements to maintain spent fuel integrity. In 2010, the Commission issued an update to its Waste Confidence Decision and Rule discussed in 10 CFR 51.23, "Temporary Storage of Spent Fuel after Cessation of Reactor Operation -- Generic Determination of No Significant Environmental Impact." The Commission concluded that spent fuel can be stored safely in SFPs or in onsite independent spent fuel storage installations without significant environmental impacts for at least 60-years beyond the plant's licensed life (which may include the term of a renewed license). In June 2012, the U.S. Court of Appeals for the DC Circuit found that some aspects of the 2010 Waste Confidence Decision and Rule did not satisfy the NRC's obligations under the National Environmental Policy Act. In response to the U.S. Court of Appeals' decision, the Commission is currently updating its Waste Confidence Decision and Rule, with completion expected in September 2014.

6.3.10 Reactor Safety Research Program

The NRC conducts reactor safety research to support its mission of ensuring that its licensees safely design, construct, and operate light water nuclear reactor facilities. The agency carries out this research program to (1) identify, evaluate, and resolve safety issues, (2) ensure that an independent technical basis exists to review licensee submittals, (3) evaluate operating experience and results of risk assessments for safety implications, and (4) support the development and use of risk-informed regulatory approaches. The NRC has an office dedicated to agency research activities that plays a similar role to a technical support organization in other countries. In conducting the Reactor Safety Research Program, the NRC anticipates challenges that the introduction of new technologies poses. The NRC also continues to seek out opportunities to leverage its resources through domestic and international cooperative research programs with other U.S. government agencies, industry organizations, and international regulatory counterparts and technical support organizations. The NRC is careful to maintain its independence and not cede its regulatory decisionmaking role to any external entities. The NRC also continues to provide opportunities for stakeholder involvement and feedback on its research program.

The NRC Reactor Safety Research Program also supports the agency's pre-application reviews for advanced non-light-water reactor designs. In the pre-application phase, the NRC interacts with prospective design certification applicants to address topics that would benefit both the applicant and the staff in preparing for a design certification application. The Commission's Policy Statement on Advanced Reactors (SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," dated April 2,

1993) encourages early interactions on such advanced designs to facilitate the resolution of safety issues early in the design process. In addition, the agency will conduct research to address technical issues that it anticipates will arise during its review of advanced reactor designs.

6.3.11 Public Participation

The NRC views nuclear regulation as the public's business. As such, the agency believes that nuclear regulation should be transacted as openly and candidly as possible to maintain and enhance the public's confidence. Ensuring appropriate openness explicitly recognizes that the public must be informed about, and have a reasonable opportunity to participate meaningfully in, the NRC's regulatory processes.

The NRC extends opportunities to participate in the agency's regulatory process to a diverse body of stakeholders, including the general public, Congress, other Federal agencies, States, local governmental bodies, Indian tribes, industry, technical societies, the international community, and citizen groups. Numerous NRC programs and processes provide the public with accessibility to NRC staff and other resources; seek to make communication with stakeholders more clear, accurate, reliable, objective, and timely; and help to ensure that the reporting of nuclear power plants' performance is open and objective. The agency has developed Web sites, and has used other electronic social media (e.g., Twitter and the NRC blog), to disseminate timely, accurate information on issues of interest to the public or events at nuclear facilities. The NRC elicits public involvement early in the regulatory process to address any safety concerns in a timely manner. In addition to the formal petition and hearing processes integrated into the licensing program, the agency also uses feedback forms at public meetings to obtain public input. Section 7.2.2 of this report provides more information about the NRC's hearing process. Section 8.1.7 provides more information on the tools that NRC uses to ensure openness and transparency in its work.

The NRC manages its rulemaking dockets using the Federal Docket Management System, a tool that provides a single point of access at http://www.regulations.gov across the Federal Government. Through this Web site, the public can access more than 11,300 documents related to almost 850 rulemaking actions that the NRC conducted from May 1996 through January 2013. The documents featured on this Web site include public comments, petitions for rulemaking, *Federal Register* notices, and their supporting materials. The public is also able to search the NRC's official records with ease by using the NRC's Agencywide Documents Access and Management System (ADAMS), which can be accessed at http://adams.nrc.gov/wba/.

Fostering an environment in which safety issues can be identified openly without fear of retribution is of paramount importance to the NRC. The agency has established tools for the public, industry, and NRC employees to use to raise safety concerns, including the NRC's petition process under 10 CFR 2.206, "Requests for Action under this Subpart;" Safety Conscious Work Environment guidance documents and related regulatory programs; and the Allegation Program.

The NRC's petition process regulations in 10 CFR 2.206 allow any member of the public to raise potential health and safety concerns and ask the agency to take specific enforcement actions against an NRC licensee. If warranted, the NRC can modify, suspend, or revoke a license, or take other appropriate enforcement action, to resolve a problem identified in the petition. The NRC's procedures governing this petition process emphasize a timely response to the petitioner and encourage increased, direct involvement of the petitioner (in addition to involvement of the licensee) by allowing the petitioner to address the petition review board personally and comment on the agency's decision.

Additionally, any member of the public may petition the NRC to develop, change, or rescind a rule under 10 CFR 2.802, "Petition for Rulemaking." Upon receiving the petition, the NRC evaluates whether the petition meets the threshold requirements in 10 CFR 2.802(c). If it does, the NRC dockets the petition and assigns it a docket number. If the petition does not meet the threshold requirements, the NRC sends a letter to the petitioner explaining why the petition does not meet those requirements. The NRC generally publishes a notice of receipt of a petition in the *Federal Register* that invites public comment (typically, a 75-day period). In some instances, the NRC determines that the issues the petitioner raised should be considered in rulemaking and publishes a notice of consideration instead of a notice of receipt. The NRC staff will evaluate the petition and any comments received and may either determine to consider the petition in a current or future rulemaking or deny the petition (in its entirety or in part). If the NRC decides to consider the petition, it publishes a proposed rule that addresses the issues raised in the petition. This action is followed by a public comment period and publication of a final rule. If the NRC denies a petition, the NRC publishes a notice of denial in the *Federal Register*. This notice of denial addresses any public comments received on the petition and the reason for denying the petition.

In addition to these formal processes, the NRC encourages workers in the nuclear industry to take their concerns directly to their employers. The agency is vigilant about fostering a safety-conscious work environment both within the NRC and within the nuclear industry that encourages reporting of safety and regulatory issues. The NRC expects licensees and other employers subject to NRC authority to establish and maintain a work environment in which employees do not fear retribution by a licensee for raising concerns about safety or regulatory issues. Within the NRC, the agency emphasizes the importance of fostering and maintaining an open, collaborative work environment that encourages all NRC employees and contractors to promptly share concerns and differing views without fear of negative consequences. These expectations are communicated through the NRC's Safety Culture Policy Statement, Safety Conscious Work Environment guidance documents, and other related regulatory tools such as safety culture case studies. Additionally, workers and members of the public may bring their concerns about safety or regulatory issues directly to the NRC. The agency established a toll-free safety hotline for reporting such concerns. NRC management, staff, and inspectors, including the resident inspectors at plant sites, are trained and available to receive such concerns. Workers and members of the public also may report concerns by email to the NRC's Allegation Program.

Historically, industry workers or members of the public report approximately 600 potential allegations directly to the NRC Allegation Program each year. The NRC developed the Allegation Program to establish a formal process for evaluating and responding to each issue. The program's primary purpose is to provide an alternative method for individuals to raise safety or regulatory issues and have them addressed. About 70 percent of the issues reported to the NRC are from licensee employees, employees of contractors to licensees, or former employees of licensees or contractors. The NRC staff will evaluate each issue to determine whether it can verify the issue and, if so, the effect of the issue on plant safety. This evaluation process involves an engineering review, inspection, or investigation by the NRC staff, or an evaluation by the licensee that is independently assessed by the NRC staff. Historically, the NRC has been able to substantiate about 30 percent of the allegations received. If the evaluation reveals a violation of regulatory requirements, the agency takes appropriate enforcement action.

Additionally, the NRC informs, in writing, the individual who raised the issue of the results of its evaluation, except in limited instances when sensitive security-related matters are involved.

6.4 Fukushima Lessons Learned

The flexibility of the existing NRC regulatory processes has enabled the United States to effectively implement lessons learned from the accident. The NRC has the authority to take necessary actions to protect public health and safety, and may demand immediate licensee response, including plant shutdowns, if necessary. The NRC took immediate action following the Fukushima accident, in the form of orders, specific inspection procedures (known as temporary instructions (TIs)), information notices (INs), and bulletins to confirm and ensure that there were no safety concerns at American nuclear facilities. Because no imminent safety issue existed, no nuclear power plants in the United States were shut down as a result of the accident in Japan.

The NRC continues to implement Fukushima lessons learned within existing regulatory processes that include review of industry response to orders, requests for information (RFIs), use of operating experience, rulemaking, and conducting additional research.

ARTICLE 7. LEGISLATIVE AND REGULATORY FRAMEWORK

1. Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.

2. The legislative and regulatory framework shall provide for:

(i) the establishment of applicable national safety requirements and regulations

(ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a license

(iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses

(iv) the enforcement of applicable regulations and of the terms of licenses, including suspension, modification, or revocation

This section explains the legislative and regulatory framework governing the U.S. nuclear industry. It discusses the provisions of that framework for establishing national safety requirements and regulations and systems for licensing, inspection, and enforcement. It also addresses lessons learned from the Fukushima accident.

7.1 Legislative and Regulatory Framework

The Atomic Energy Act of 1954, passed by Congress and signed into law by the President, established the Atomic Energy Commission and the legal framework for all subsequent regulation of nuclear installations. However, as is generally the case with most laws, this act provided general principles and concepts and left the regulatory body (now the U.S. Nuclear Regulatory Commission, or NRC) to address the details through specific regulations. The Energy Reorganization Act of 1974, likewise passed by Congress and signed into law by the President, abolished the Atomic Energy Commission and created the NRC to regulate commercial nuclear activities and the U.S. Energy Research and Development Administration (ERDA) to continue Government-sponsored nuclear activities. ERDA was subsequently incorporated into the U.S. Department of Energy (DOE). The Administrative Procedure Act provides the general rules and procedures through which the Atomic Energy Act is implemented.

The United States has also ratified various international conventions that affect nuclear safety:

- The Treaty on the Non-Proliferation of Nuclear Weapons, ratified in 1970, governs the NRC's export licensing activities.

- The U.S.-IAEA Safeguards Agreement, ratified in 1980, requires eligible facilities in the United States to report material accounting data on declared nuclear material. The Agreement further requires eligible facilities to submit to IAEA inspections. The Additional Protocol to the US-IAEA Safeguards Agreement, ratified in 2004, strengthened IAEA reporting and access rights for eligible facilities.

- The Convention on the Physical Protection of Nuclear Material, ratified in 1982, requires NRC licensees to take steps to protect nuclear material during international transport.

- The Convention on Early Notification of a Nuclear Accident, ratified in 1988, requires the NRC to help the U.S. Department of State report significant accidents to IAEA and any State affected by a transboundary radioactive release.

- The Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, ratified in 1988, requires the NRC to help the U.S. Department of State respond to requests for assistance in the event of a foreign nuclear accident or emergency.

- The Convention on Nuclear Safety (CNS), ratified in 1999, calls for periodic review meetings of all the Contracting Parties. Before the review meeting, the CNS requires the United States to submit a National Report that details the United States' commitment to nuclear safety.

- The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management ("Joint Convention"), ratified in 2003, requires the United States to take steps to ensure that individuals and the environment are protected against radiological hazards at all stages of radioactive waste and spent fuel management. The Joint Convention further calls for periodic review meetings of all the Contracting Parties. Before the review meeting, each Contracting Party must submit a national report that addresses measures taken to implement the obligations under the Joint Convention.

- The Convention on Supplementary Compensation for Nuclear Damage, ratified in 2008, requires the United States to ensure that adequate compensation exists in the event that "nuclear damage" results from a nuclear incident.

7.2 Provisions of the Legislative and Regulatory Framework

7.2.1 National Safety Requirements and Regulations

In addition to the Atomic Energy Act, several statutes (listed in previous U.S. National Reports and briefly described in Section 8.1.2.1) have substantial bearing on the Commission's practices and procedures. Furthermore, various U.S. Presidents have issued executive orders and directives that affect nuclear safety. For example, President Reagan issued Executive Order 12656, "Assignment of Emergency Preparedness Responsibilities," on November 18, 1988. This Executive Order assigned certain emergency preparedness responsibilities to the NRC in case of a national emergency. Likewise, in the wake of the Three Mile Island accident, President Carter directed FEMA to direct all offsite emergency activities and review emergency plans in States with operating reactors. As another example, the NRC has voluntarily complied with President Clinton's Executive Order 12898, "Federal Actions To Address Environmental Justice in Minority Populations and Low-Income Populations," dated February 11, 1994, which requires Federal agencies to consider whether their programs or policies have a disproportionately adverse health or environmental effect on minority populations.

The NRC has implemented these statutes and executive orders through regulation and guidance. Specifically, Title 10 of the *Code of Federal Regulations* (10 CFR), "Energy," governs, among other things, the licensing of nuclear installations. The NRC established these regulations through "notice-and-comment" rulemaking procedures under the Administrative Procedure Act. In short, these rulemaking procedures include: (1) establishing a technical or legal basis, or both, for the proposed rule, (2) publishing a proposed rule for public comment, and (3) after considering comments, issuing a final rule. Once these final rules are in place, they are binding on the regulated entities (including operators of nuclear installations) and can be revised only through a new notice-and-comment rulemaking. This ensures that interested parties remain both informed of, and involved with, any changes to the NRC's regulatory scheme.

7.2.2 Licensing of Nuclear Installations

The NRC must license all commercial nuclear installations (e.g., nuclear power plants) in the United States. (As discussed in Section 8.1.2.1, some Federal Government facilities that are operated by or for DOE are not subject to NRC licensing under the Atomic Energy Act and the Energy Reorganization Act except where specifically provided by law). The Atomic Energy Act, Chapter 10, Section 101, prohibits possession and operation of a nuclear installation without a valid license. Sections 101 and 103 further provide that only the NRC is authorized to issue a license for nuclear reactor facilities. Section 103 also states that such licenses are subject to conditions that the NRC may establish by rule or regulation to carry out the purposes and provisions of the Atomic Energy Act.

The Atomic Energy Act, Section 189a, provides interested parties with hearing rights in proceedings for the granting, suspending, revoking, or amending of licenses or construction permits for production and utilization facilities. Hearings are conducted under procedural rules stated in 10 CFR Part 2, "Agency Rules of Practice and Procedure," and, in particular, Subpart C, "Rules of General Applicability: Hearing Requests, Petitions to Intervene, Availability of Documents, Selection of Specific Hearing Procedures, Presiding Officer Powers, and General Hearing Management for NRC Adjudicatory Hearings," in conjunction with the subpart of 10 CFR Part 2 that governs the particular proceeding. The NRC staff participates as a party in almost all hearings. Hearings are usually held before a three-member Atomic Safety and Licensing Board, which is generally composed of one lawyer and two technical members but also may be conducted by a single licensing board member (i.e., presiding officer) or the Commission.

Two alternative approaches for NRC licensing of nuclear reactor facilities exist. The original licensing approach, under 10 CFR Part 50, requires two steps. In the first step, the NRC reviews a preliminary application and decides whether to grant a construction permit. In the second step, the agency reviews the final application and decides whether to grant an operating license. The NRC licensed all current operating nuclear power plants in the United States according to this process.

In 1989, the Commission established an alternative licensing system, published in 10 CFR Part 52, which provides for certified standard designs and combined licenses that resolve design issues before construction, and early site permits that resolve most siting issues years before construction. The basic concept underlying 10 CFR Part 52 is to provide for standardization and early resolution of licensing issues by approving nuclear reactor designs through generic rulemaking. Once the designs are approved (i.e., certified), an applicant can reference them in applications for permission to build and operate nuclear power plants without needing to relitigate, in individual hearings, the issues resolved in the rulemaking.

The 10 CFR Part 52 process requires the NRC to determine and approve, before construction, the criteria that will be used to evaluate, after construction, whether the plant has been built as specified in the design. Before authorizing operation, the Commission must determine that these criteria have been met. The determination of whether a specific plant meets the acceptance criteria is subject to hearing rights.

Once licensed, a nuclear power plant can renew its operating license for up to an additional 20 years. The NRC provides the licensing system for license renewal under 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and interested parties have hearing rights under 10 CFR Part 2 in renewal proceedings.

7.2.3 Inspection and Assessment

Under the Atomic Energy Act, the NRC has the authority to inspect nuclear power plants in its role of protecting public health and safety and the common defense and security. The NRC staff inspects power reactors under construction, in test conditions, and in operation to ascertain compliance with regulations and license conditions. Through its inspection program, the NRC assesses whether activities are properly conducted and equipment is properly maintained to ensure safe operations. The agency integrates inspection results into its overall evaluation of licensee performance, as discussed in Article 6 of this report. As described in the next section, the NRC may take enforcement action to address violations of NRC requirements.

A feature of the NRC's inspection program is the assignment of resident inspectors to nuclear power plants. At least two inspectors are assigned to each nuclear power site, and these inspectors continuously monitor licensee activities in accordance with the NRC's baseline inspection program. To supplement these continuous inspections, regional inspection specialists conduct periodic inspections of each plant in his or her region. If needed, regional inspectors perform special investigations of plants that exceed established thresholds during routine inspections and thus require heightened scrutiny. All inspection findings are recorded, and the NRC typically issues inspection reports for a specific power plant quarterly. Additionally, senior agency managers review plants that have performance issues and report these results to the Commission during the annual Agency Action Review Meeting. This meeting provides another opportunity to discuss significant events, licensee performance issues, trends, and actions to mitigate recurrences.

7.2.4 Enforcement

The NRC draws its jurisdiction for enforcement from the Atomic Energy Act and the Energy Reorganization Act.

The Atomic Energy Act, Section 161, authorizes the NRC to conduct inspections and investigations and to issue orders as may be necessary or desirable to promote the common defense and security, protect health, or minimize danger to life or property. Section 186 authorizes the NRC to revoke licenses under certain circumstances (e.g., for material false statements, for a change in conditions that would have warranted NRC refusal to grant a license on an original application, for a licensee's failure to build or operate a facility in accordance with the terms of the permit or license, and for a violation of an NRC regulation). Section 234 authorizes the NRC to impose monetary civil penalties not to exceed $100,000 per violation per day; however, that amount is adjusted every 4 years by the Federal Civil Penalties Inflation Adjustment Act of 1990, as amended by the Debt Collection Improvement Act of 1996, and is currently $140,000. In addition to the provisions mentioned in Section 234, Sections 84 and 147 authorize the imposition of civil penalties for violations of the regulations that implement those provisions. Section 232 authorizes the NRC to seek injunctive or other equitable relief for violations of regulatory requirements.

The Atomic Energy Act, Chapter 18, provides for varying levels of criminal penalties (i.e., monetary fines and imprisonment) for willful violations of the Act, or of regulations or orders issued by the NRC under Sections 65, 161b, 161i, or 161o of the Act. Section 223 allows the NRC to impose criminal penalties on certain individuals who are employed by firms constructing or supplying basic components of any utilization facility if the individual knowingly and willfully violates NRC requirements in a way that could significantly impair a basic component. Section 235 allows the NRC to impose criminal penalties on persons who interfere with nuclear inspectors. Section 236 allows the NRC to impose criminal penalties on persons who cause, or attempt to cause, sabotage at a nuclear facility or to nuclear fuel. The agency refers alleged or suspected instances of criminal violations of the Atomic Energy Act to the U.S. Department of Justice for appropriate action.

The Energy Reorganization Act, Section 206, authorizes the NRC to impose civil penalties on licensees and individuals or responsible persons for knowing and consciously failing to provide the agency with certain safety information.

Subpart B, "Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties," of 10 CFR Part 2 specifies the procedures that the NRC uses in exercising its enforcement authority. The scope of Subpart B includes the following procedures:

- 10 CFR 2.201, "Notice of Violation," outlines the procedure for issuing notices of violations.

- 10 CFR 2.202, "Orders," explains the procedure for issuing orders. In accordance with this section, the NRC may decide to issue an order to institute a proceeding to modify, suspend, or revoke a license or to take other action against an NRC licensee or other person subject to the NRC's jurisdiction. The licensee or any other person adversely affected by the order may request a hearing. The NRC is authorized to make orders immediately effective if necessary to protect public health, safety, or interest, or if the violation is willful.

- 10 CFR 2.204, "Demand for Information," specifies the procedure for issuing a demand for information to a licensee or other person subject to the NRC's jurisdiction to determine whether an order should be issued or other enforcement action should be taken. Because the agency is only seeking information, demands for information are not subject to hearing rights. A licensee must answer a demand for information. An unlicensed person may answer a demand either by providing the requested information or by explaining why the NRC should not have issued the demand.

- 10 CFR 2.205, "Civil Penalties," describes the procedure for assessing civil penalties. The NRC initiates the civil penalty process by issuing a notice of violation and proposed imposition of a civil penalty. The agency provides the person charged with the civil penalty with an opportunity to contest in writing the proposed imposition of a civil penalty. After evaluating the response, the NRC may mitigate, remit, or impose the civil penalty. If the agency imposes a civil penalty, it provides an opportunity for a hearing. If a civil penalty is not paid following a hearing, or if a hearing is not requested, the agency may refer the matter to the U.S. Department of Justice to institute a civil action in Federal district court to collect the penalty.

The NRC's enforcement process is also discussed in Section 9.3.

7.3 Fukushima Lessons Learned

The United States has not changed the legislative framework governing the U.S. nuclear industry since Fukushima. The NRC has taken some regulatory actions in response to Fukushima and continues to consider whether additional actions, such as amendments to NRC regulations, are appropriate. On March 12, 2012, the NRC issued the first regulatory requirements, in the form of orders, based on the lessons learned from Fukushima. The three orders require safety enhancements of operating reactors, construction permit holders, and combined license holders. These orders require nuclear power plants to implement safety enhancements related to: (1) mitigation strategies to respond to external events resulting in the loss of power at plants, (2) ensuring reliable hardened containment vents for BWR Mark I and II designs, and (3) enhancing spent fuel pool (SFP) instrumentation. Operating plants are required to promptly begin implementation of the safety enhancements and complete implementation within two refueling outages or by December 31, 2016, whichever comes first.

In addition, the NRC issued an RFI, requesting each reactor to reevaluate the seismic and flooding hazards at its site using present-day methods and information, conduct walkdowns of its facilities to ensure protection against the hazards in its current design-basis, and assess its emergency communications systems and staffing levels.

ARTICLE 8. REGULATORY BODY

1. Each Contracting Party shall establish or designate a regulatory body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence, and financial and human resources to fulfill its assigned responsibilities.

2. Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.

This section explains the establishment of the U.S. regulatory body (i.e., the NRC). It also explains how the functions of the NRC are separate from those of bodies responsible for promoting research, development and advancement of nuclear energy (e.g., DOE). It discusses financial and human resources aspects, the regulatory body's international responsibilities, and its policy for maintaining openness and transparency. Lastly, this section addresses lessons learned from Fukushima.

8.1 <u>The Regulatory Body</u>

This section explains the NRC's mandate, authority and responsibilities, structure and position in the Government, international responsibilities and activities, such as those related to international standards and Integrated Regulatory Review Service (IRRS) and Operational Safety Assessment Review Team (OSART) missions, and its financial and human resources.

8.1.1 Mandate

As discussed in Article 7, the U.S. Congress created the NRC as an independent regulatory agency in January 1975, with the passage of the Energy Reorganization Act. In giving the NRC an exclusively regulatory mandate, the statute reflected (in part) a congressional judgment that the expanding commercial nuclear power industry (which was expected to continue to grow) warranted the full-time attention of an exclusively regulatory agency. In creating the NRC, the U.S. Congress also addressed a developing public concern that regulatory responsibilities were overshadowed by the promotion of nuclear power at the Atomic Energy Commission.

8.1.2 Authority and Responsibilities

8.1.2.1 Scope of Authority

The NRC's mission is to ensure that the civilian uses of nuclear energy and materials in the United States are conducted with proper regard for public health and safety, national security, and environmental concerns. The Atomic Energy Act provides the charter for most of these regulatory responsibilities. In the Atomic Energy Act, the U.S. Congress created a national policy of developing the peaceful uses of atomic energy. The U.S. Congress has amended the statute over the years to address developing technology and changing regulatory needs. Other, more specialized, statutes prescribe the NRC's duties with regard to high-level radioactive waste, low-level radioactive waste, mill tailings, environmental reviews, nonproliferation, antiterrorism, and import and export of nuclear materials and equipment. In addition, the National

Environmental Policy Act of 1969, as amended, imposes broad environmental responsibilities on all Federal agencies, including the NRC.

The NRC's licensing authority extends to other Government organizations (such as the Tennessee Valley Authority (TVA), which operates nuclear power plants) and to the military's use of radiopharmaceuticals in its hospitals. But the NRC's licensing authority does not extend to the military's or DOE's nuclear weapons programs and facilities, nor to DOE's test and research reactors. Section 8.2 of this report provides specific information on the scope of the agency's limited jurisdiction over DOE nuclear installations. The NRC's responsibilities include ensuring both the safety and the security of commercial nuclear facilities and materials.

8.1.2.2 The NRC as an Independent Regulatory Agency

The Commission's status as an independent regulatory agency within the Executive Branch of the Federal Government means that the President cannot ordinarily direct the agency's regulatory decisions. There are two statutory sources of the Commission's independence from presidential direction. First, the President can remove an NRC Commissioner only for cause – namely, "inefficiency, neglect of duty, or malfeasance in office." The President can, however, designate one member of the Commission as Chairman to serve as such at the pleasure of the President. Second, the Commission has the statutory right to defend itself whenever its adjudicatory or rulemaking decisions are challenged in U.S. appellate courts.

Congress cannot override the Commission's decisions, except by duly enacted legislation. The courts are likewise limited in reviewing the NRC's factual safety findings. Although a Federal appellate court can overturn a Commission decision for violations of law, safety findings will generally be overturned only if they are arbitrary. This provides the Commission with some degree of independence from the Judiciary.

The independence of the NRC's decisionmaking process implies a responsibility on the part of the Commissioners and their personal staff to keep the process free from improper outside influence. This is especially important in the case of adjudications. When the Commissioners take part in adjudications, they ordinarily act in the role of appellate judges (reviewing the decisions of Atomic Safety and Licensing Board judges) and, in general, are bound by the same kinds of strictures that apply to Federal court judges.

8.1.3 Structure of the Regulatory Body

This section explains the structure of the NRC. It covers the Commission, component offices and their responsibilities, and advisory committees and their functions. It also explains recent changes in NRC organization.

8.1.3.1 The Commission

The NRC is headed by a five-member Commission appointed by the President and confirmed by the U.S. Senate. The President designates one member to serve as Chairman and official spokesperson. The Commission as a whole formulates policies and regulations governing safety and security, it issues orders to licensees, and adjudicates legal matters brought before it. The Executive Director for Operations carries out the policies and decisions of the Commission and directs the activities of the program offices.

8.1.3.2 Component Offices of the Commission

The following offices report directly to the Chairman or the Commission:

- Office of the Executive Director for Operations. The Executive Director for Operations is the chief operational and administrative officer of the Commission and is authorized and directed to discharge licensing, regulatory, and administrative functions, as well as other actions necessary for day-to-day agency operations. The Executive Director for Operations supervises and coordinates the policy development and operational activities of the NRC program and regional offices, and implements Commission policy directives pertaining to these offices.

- Office of the Chief Financial Officer. The Office of the Chief Financial Officer leads the agency in planning, acquiring and ensuring the appropriate use of financial resources, and provides financial services to support the agency's mission.

- Office of Commission Appellate Adjudication. The Office of Commission Appellate Adjudication is responsible for assisting the Commission in the exercise of its quasi-judicial functions, including the resolution of appeals from decisions of the Atomic Safety and Licensing Boards. This office provides the Commission with an analysis of any appellate adjudicatory matter that may merit a Commission decision, and drafts adjudicatory decisions under the Commission's guidance.

- Office of Congressional Affairs. The Office of Congressional Affairs is the primary point of contact for all communications between the NRC and Congress. This office provides advice and assistance to the Chairman, the Commission, and NRC staff on congressional matters; monitors legislative proposals, bills, and hearings; informs the NRC of the views of Congress on NRC policies, plans, and activities; provides timely responses to congressional requests for information; and provides the information necessary to keep appropriate members of Congress and congressional staff fully and currently informed of NRC actions.

- Office of the General Counsel. The Office of the General Counsel directs matters of law and legal policy, providing opinions, advice, and assistance to the agency on all of its activities.

- Office of International Programs. The Office of International Programs coordinates the NRC's international activities and provides assistance and recommendations to the Chairman, the Commission, and the NRC staff on international outreach activities. It plans, develops, and implements programs to carry out policies in the international arena, including export and import licensing responsibilities. It establishes and maintains working relationships with individual countries and international nuclear organizations, as well as other involved U.S. Government agencies.

- Office of Public Affairs. The Office of Public Affairs directs the agency's public affairs program, advising agency officials and developing key strategies that help increase public confidence in NRC policies and activities. This includes keeping top management informed of public interest in and news coverage of the NRC's regulatory activities, as well as providing timely, clear, and accurate information on NRC activities to the public and the media who call or email the agency and through news releases, fact sheets, brochures, interviews, Web postings, and social media.

- Office of the Secretary of the Commission. The Office of the Secretary of the Commission provides executive management services to support the Commission and to carry out Commission decisions. It assists with the planning, scheduling, and conduct of Commission business; maintains historical paper files of official Commission records; administers the NRC Historical Program; and maintains the Commission's official adjudicatory and rulemaking dockets.

8.1.3.3 Offices of the Executive Director for Operations

The offices reporting to the Executive Director for Operations ensure that the commercial use of nuclear materials in the United States is safely conducted. The NRC offices are briefly described below.

- Office of Enforcement. The Office of Enforcement oversees, manages, and directs the development and implementation of policies and programs for enforcing NRC requirements. It oversees the agency's allegations management program and the allegations review process. The office is responsible for safety culture policy matters, the agency's Alternative Dispute Resolution Program related to enforcement matters, and the agency's internal Differing Views Program.

- Office of Information Services. The Office of Information Services plans, directs, and oversees the delivery of centralized information technology infrastructure, applications, and information management services, in addition to the development and implementation of information technology and management plans, architecture, and policies to support the mission, goals, and priorities of the agency.

- Office of Investigations. The Office of Investigations develops policy, procedures, and quality control standards for investigations of licensees and applicants, as well as their contractors or vendors. This office conducts investigations of allegations of wrongdoing by non-NRC employees and contractors. The Office of Investigations is independent and may self-initiate investigations when a person or entity under its jurisdiction is suspected to have committed a matter of wrongdoing. This office plans, conducts, and makes referrals of substantiated criminal cases to the U.S. Department of Justice. This office conducts liaison with Federal, State, and local law enforcement and provides investigative assistance to NRC staff on regulatory matters. Additionally, it keeps the Commission and NRC offices apprised of regulatory matters under investigation as they affect public health and safety, the common defense and security, and the environment.

- Office of Federal and State Materials and Environmental Management Programs. The Office of Federal and State Materials and Environmental Management Programs is responsible for the safe and secure use of source, byproduct, and special nuclear materials in industrial, medical, academic, and commercial activities, and at decommissioning, uranium recovery, and low-level waste sites. It ensures effective communications and working relationships between the NRC and other governmental entities and administers the Agreement State Program (through which States have signed formal agreements with the NRC to assume regulatory responsibility over certain byproduct, source, and small quantities of special nuclear materials). It also develops and implements rules and guidance for these activities.

- Office of New Reactors. The Office of New Reactors is responsible for accomplishing key components of the NRC's nuclear reactor safety mission for new commercial reactor facilities licensed in accordance with 10 CFR Part 52 and small modular reactor and advanced reactor facilities. As such, the office conducts regulatory activities in the primary program areas of siting, licensing, and oversight of construction for new commercial nuclear power reactors.

- Office of Nuclear Material Safety and Safeguards. The Office of Nuclear Material Safety and Safeguards is responsible for regulating activities that provide for the safe and secure production of nuclear fuel used in commercial nuclear reactors; the safe storage, transportation, and disposal of high-level radioactive waste and spent nuclear fuel; and the transportation of radioactive materials regulated under the Atomic Energy Act.

- Office of Nuclear Reactor Regulation. The Office of Nuclear Reactor Regulation is responsible for accomplishing key components of the NRC's nuclear reactor safety mission to protect public health and safety and the environment. To do so, the office conducts a broad range of regulatory activities in the four primary program areas of rulemaking, licensing, oversight, and incident response for commercial nuclear power reactors and test and research reactors.

- Office of Nuclear Regulatory Research. The Office of Nuclear Regulatory Research plans, recommends, and conducts research programs and technical safety reviews that support the resolution of ongoing and future safety issues identified as regulatory needs by offices with regulatory functions or through its own long-term research program.

- Office of Nuclear Security and Incident Response. The Office of Nuclear Security and Incident Response develops overall agency policy and provides management direction for evaluating and assessing technical issues involving security, safeguards and emergency preparedness at nuclear facilities. The office is the agency's safeguards, security, emergency preparedness and incident response interface with other Federal agencies.

- Regional Offices. The four regional offices conduct inspections, and execute established policies related to licensing and construction, allegation, enforcement, emergency response, and Government liaison programs in the U.S.-licensed nuclear facilities. The regional offices also manage decommissioning activities.

Supporting the Executive Director for Operations are the Offices of Administration, Human Resources, Small Business and Civil Rights, and Computer Security:

- Computer Security Office. The Computer Security Office plans, directs, and oversees the implementation of a comprehensive, coordinated, integrated and cost-effective NRC information technology security program. The office provides cyber security oversight for agency systems; and adjusts NRC's cyber security program to counter the evolving threat to electronic information in accordance with applicable laws, regulations, Commission guidance, Chief Information Officer direction, and NRC policies and management initiatives.

- Office of Administration. The Office of Administration provides centralized services in the areas of contracts, facilities and security, property management, and administration, including support for rulemaking and agency directives, transportation, parking, translations, audiovisual needs, food services, mail distribution, labor services, furniture and supplies, and other areas.

- Office of the Chief Human Capital Officer. The Office of the Chief Human Capital Officer provides overall management of the agency's human capital planning and training and development programs. Accordingly, this office is responsible for implementing human resource policy and operations agencywide. This includes overseeing the development and implementation of human resources management and information systems for staffing, strategic workforce planning, and other corporate activities to support a skilled and dynamic workforce. The office's training and development programs are designed to establish, maintain, and enhance the skills employees need today and to meet the agency's future skill needs.

- Office of Small Business and Civil Rights. The Office of Small Business and Civil Rights is responsible for facilitating equal employment opportunity for all NRC employees, applicants for employment, and business partners through an ongoing affirmative employment and diversity management process, implementation of civil rights statutes, execution of outreach and compliance coordination mandates, and employment of maximum small business participation in acquisitions.

8.1.3.4 Advisory Committees

The three principal advisory committees for NRC programs are the Advisory Committee on Reactor Safeguards, the Advisory Committee on the Medical Uses of Isotopes and the Committee to Review Generic Requirements. In addition, the NRC has established an ad hoc Licensing Support Network Advisory Review Panel.

- Advisory Committee on Reactor Safeguards. The Advisory Committee on Reactor Safeguards reviews and reports on safety studies and reactor facility license and license renewal applications, advises the Commission on the hazards of proposed and existing reactor facilities and the adequacy of proposed reactor safety standards, advises the Commission on issues associated with nuclear materials and waste management, initiates reviews of specific generic matters or nuclear facility safety-related items, and reviews the NRC's research activities.

- Advisory Committee on the Medical Uses of Isotopes. The Advisory Committee on the Medical Uses of Isotopes advises the NRC on policy and technical issues that arise in the regulation of the medical uses of radioactive material in diagnosis and therapy.

- Committee to Review Generic Requirements. The Committee to Review Generic Requirements ensures that proposed generic backfits to be imposed on NRC-licensed power reactors and selected nuclear materials licensees are appropriately justified, based on the backfit provisions of applicable NRC regulations and the Commission's backfit policy.

8.1.3.5 Atomic Safety and Licensing Board Panel

In addition to the advisory committees, the NRC has an Atomic Safety and Licensing Board Panel. Administrative judges and administrative law judges who are members of this panel —either as a single presiding officer or in three-judge boards— conduct hearings for the Commission. Additionally, the panel performs such other regulatory functions as the Commission authorizes. The panel's Chief Administrative Judge develops and applies procedures governing the activities of boards, administrative judges, and administrative law judges. The Chief Administrative Judge also makes appropriate recommendations to the Commission concerning the rules governing the conduct of hearings.

8.1.3.6 Office of the Inspector General

The Inspector General provides leadership and policy direction in conducting audits and investigations to promote economy, efficiency, and effectiveness within the NRC and to prevent and detect fraud, waste, abuse, and mismanagement in agency programs and operations. The Inspector General recommends corrective actions to be taken, reports on progress made in implementing those actions, and reports criminal matters to the U.S. Department of Justice. The Inspector General serves under the general supervision of the NRC Chairman but operates with personnel, contracting, and budget authority independent of that of the NRC.

8.1.4 Position of the NRC in the Governmental Structure

This section explains the relationship of the NRC to the Executive Branch, the States, and Congress.

8.1.4.1 Executive Branch

The components of the Executive Branch with which the NRC has the most frequent contact and interaction are the White House, Office of Management and Budget (OMB), U.S. Department of State, DOE, U.S. Environmental Protection Agency, U.S. Department of Homeland Security (DHS), Federal Emergency Management Agency (FEMA), U.S. Department of Labor, U.S. Department of Transportation, and U.S. Department of Justice. Section 8.2 of this report discusses the NRC's relationship to DOE. The following summarizes the agency's relationships with the other identified components of the Federal Government:

- The White House. As noted in Section 8.1.2.2, as an independent regulatory agency, the White House cannot directly set NRC policy. It may, however, influence NRC policy by (1) appointing Commissioners and a Chairman in whose outlook and judgment it has confidence and (2) making its views known on nonadjudicatory matters. In certain areas, such as national security policy, the Commission has declared its intent to give great weight to the views of the Executive Branch. In informal policy matters, such as rulemaking, White House and Executive Branch officials may properly try to influence NRC decisions. Ultimately, however, the NRC must make the decision and accept responsibility for it.

- Federal Emergency Management Agency. FEMA assists the NRC's licensing process by preparing reviews and evaluations and by presenting witnesses to testify at licensing hearings. FEMA also participates with the NRC in observing and evaluating emergency exercises at nuclear plants. FEMA findings are not binding on the NRC, but they are presumed to be valid unless controverted by more persuasive evidence. FEMA is part of DHS.

- U.S. Department of Homeland Security. The NRC routinely interfaces with DHS regarding infrastructure protection and cyberspace issues. The mission of DHS is to secure the nation from threats.

- U.S. Department of Justice. As mentioned in Section 8.1.2.2, the NRC has independent litigation authority, which allows it to defend itself in U.S. appellate courts. However, under the Administrative Orders Review Act (commonly called the Hobbs Act), the United States is a party to petitions for review challenging NRC licensing decisions or regulations. Thus, NRC litigation almost always requires coordination with the U.S. Department of Justice.

  In addition, the NRC's Office of Investigations investigates alleged wrongdoing by NRC licensees, certificate holders, permit holders, or applicants; contractors, subcontractors, and vendors of such entities; and employees of these entities who may have committed violations of the Atomic Energy Act or the Energy Reorganization Act. All substantiated criminal cases are referred to the U.S. Department of Justice for prosecution consideration.

  The NRC's Office of the Inspector General provides information to the Department of Justice whenever it has reasonable grounds to believe that an NRC employee or contractor has violated Federal law. The Inspector General refers cases for review for possible criminal prosecution to the U.S. Attorney's Office for the area in which the potential violation occurred. When the Department of Justice desires support from the Office of the Inspector General for investigations or grand jury work, it makes the request directly to the Inspector General.

- U.S. Department of Labor. The NRC monitors discrimination actions related to NRC-licensed activities filed with the U.S. Department of Labor under Section 211 of the Energy Reorganization Act. The NRC also develops enforcement actions when there are properly supported findings of discrimination, either from the NRC's Office of Investigations or from U.S. Department of Labor adjudications.

- U.S. Department of State. By law, the NRC licenses the export and import of commercial nuclear equipment and material. For significant license applications, the Commission requests the U.S. Department of State to provide Executive Branch views on whether the license should be issued.

  The NRC works with the U.S. Department of State during negotiation of international agreements in the nuclear field and coordinates all interactions with IAEA and other international organizations of the United Nations, as well as the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development. In general, these interactions serve to develop policy on international nuclear issues that are under NRC domestic purview and to plan and coordinate programs of nuclear safety and safeguards assistance to other countries.

- U.S. Department of Transportation. The NRC and the U.S. Department of Transportation share responsibility for the control of radioactive material transport. U.S. Department of Transportation regulations cover all aspects of transportation, including packaging, shipping and carrier responsibilities, and related documentation.

- U.S. Environmental Protection Agency. The responsibilities of the NRC and the U.S. Environmental Protection Agency intersect or overlap in areas in which the U.S. Environmental Protection Agency issues generally applicable environmental standards for activities that are also subject to NRC licensing. Examples include standards for high-level waste repositories, decommissioning standards, and standards for public and worker protection. The U.S. Environmental Protection Agency has the ultimate authority to establish generally applicable environmental standards to protect the environment from radioactive material.

- U.S. Office of Management and Budget. The NRC submits its annual budget requests, including proposed personnel ceilings, to the OMB for approval.

8.1.4.2 The States (i.e., of the United States)

As explained in Article 7, the Atomic Energy Act confers on the NRC preemptive authority over health and safety regulation of nuclear energy and Atomic Energy Act materials. As a result, the general rule is that nuclear power plant safety, like airline safety, is the exclusive province of the Federal Government and cannot be regulated by the States. The courts would thus void a State law that attempted to set nuclear safety standards. However, the courts will not overturn a State law that regulates nuclear energy for purposes other than health and safety, such as economics, unless it conflicts with an NRC requirement. Similarly, the courts will not ordinarily question a State's declared purpose in enacting legislation.

However, the Atomic Energy Act did not entirely exclude States from the regulation of certain nuclear matters. Section 274 of the Act created the Agreement State Program, under which the NRC may relinquish its authority over most nuclear materials to those States willing to assume that authority. The NRC may not relinquish authority over such facilities as reactors, fuel reprocessing and enrichment plants, imports and exports, critical mass quantities of special nuclear material, high-level waste disposal, or certain other excepted areas.

Thirty-seven States have signed formal agreements with the NRC and have assumed regulatory responsibility over certain byproduct, source, and small quantities of special nuclear materials. Agreement States receive no Federal funding to support their regulatory programs. The NRC conducts performance-based reviews of Agreement State programs to ensure that they remain adequate to protect public health and safety and are compatible with the NRC materials program.

Some States have shown a desire to participate in matters relating to nuclear power plants. In response, the NRC issued a policy statement in February 1989 declaring its intent to cooperate with States in the area of nuclear power plant safety by keeping States informed of matters of interest to them and considering proposals for State officials to participate in NRC inspection activities, in accordance with a memorandum of understanding between the State and the NRC. The policy statement makes clear that States must channel their contacts with the NRC through a single State Liaison Officer, whom the Governor appoints. States are authorized only to observe and assist in NRC inspections of reactors; they cannot conduct their own independent health and safety inspections.

The NRC works in cooperation with Federal, State, and local governments; interstate organizations; and Native American Tribal Governments to maintain effective relations and communications with these organizations and to promote greater awareness and mutual understanding of the policies, activities, and concerns of all parties involved as they relate to radiological safety at NRC-licensed facilities.

8.1.4.3 Congress

The following oversight committees and subcommittees in the U.S. Senate and U.S. House of Representatives have jurisdiction over aspects of the NRC's activities. These committees and subcommittees are listed below.

- Senate Oversight. In the U.S. Senate, the Committee on the Environment and Public Works has jurisdiction over domestic nuclear regulatory activities. Within the committee, the Subcommittee on Clean Air and Nuclear Safety is responsible for regulation and oversight of the NRC. The Energy and Natural Resources Committee and the Environment and Public Works Committee share jurisdiction over nuclear waste issues.

- House Oversight. In the U.S. House of Representatives, the Committee on Energy and Commerce has jurisdiction over domestic nuclear regulatory activities. Within the committee, the Subcommittee on Energy and Power and the Subcommittee on Environment and the Economy have responsibility for regulation and oversight of the NRC.

- Other Relevant Committees. In addition to the committees and subcommittees mentioned above, the House and Senate Appropriations Subcommittees on Energy and Water Development play a key role in approving the Commission's annual budget. A number of other committees frequently interact with the NRC on international affairs, research, security, and general Governmental operations.

8.1.5 International Responsibilities and Activities

The NRC conducts international activities related to statutory mandates, international treaties and conventions, international organizations, bilateral relations, and research.

U.S. law or international treaties and conventions mandate several NRC international activities; other activities are discretionary. In particular, the NRC is statutorily mandated to serve as the U.S. licensing authority for exports and imports of nuclear materials and equipment.

The NRC supports U.S. foreign policy in the safe and secure use of nuclear materials and in guarding against the spread of nuclear weapons. The agency actively participates in developing and implementing a variety of legally binding treaties and conventions that create an international framework for the peaceful uses of nuclear energy. The NRC provides technical and legal advice and assistance to international organizations and foreign countries as they work to develop effective regulatory organizations and rigorous safety standards. Some activities are carried out within the programs of IAEA, the NEA of the Organisation for Economic Co-operation and Development, or other international organizations. The NRC conducts other activities directly with counterpart agencies in other countries under cooperation agreements.

International Treaties. Treaties that legally bind the NRC and the U.S. Government's peaceful uses of nuclear energy and nuclear applications include the 1970 Treaty on Non-Proliferation of Nuclear Weapons, the 1987 Convention on Physical Protection of Nuclear Material, the 1996 CNS, the 1986 Convention on Early Notification of a Nuclear Accident, the 1987 Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency, and the 2001 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. NRC staff regularly participate in international meetings related to these conventions and have held a variety of convention leadership positions. In its bilateral work with regulatory counterparts worldwide, the NRC seeks to exchange experience and good practices to further the goals of these international instruments.

In addition to these legally-binding obligations, the United States has agreed to comply with certain activities to enhance the safe and secure uses of nuclear applications. For example, the United States has made a political commitment to implement the IAEA Code of Conduct on the Safety and Security of Radioactive Sources. This commitment has been codified in U.S. statute as part of the Energy Policy Act of 2005 and is reflected in the NRC's export and import regulations.

Export-Import. The NRC's key international responsibility is licensing the export and import of nuclear materials and equipment for civilian use, such as low-enriched uranium fuel for nuclear power plants, high-enriched uranium for research and test reactors, nuclear reactors, certain nuclear reactor components (such as pumps and valves), and radioisotopes used in industrial, medical, agricultural, and scientific fields. The NRC ensures that such exports and imports are consistent with the goals of the safe and peaceful use of these materials and equipment, limiting the proliferation of nuclear weapons, and promoting the Nation's common defense and security. The Atomic Energy Act, the Nuclear Non-Proliferation Act of 1978, and 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material," detail the standards and procedures for issuing export and import licenses. The NRC also coordinates closely with other U.S. Government agencies, including the U.S. Department of State, U.S. Department of Commerce, and DOE, on export- or import-related matters that fall within these agencies' jurisdictions.

International Organizations and Associations. In consultation with the Executive Branch agencies, the NRC actively participates in the full scope of programs of the two major international nuclear organizations, IAEA and NEA. For example, since 1996, the United States has or is planning to participate in more than 30 OSART missions. Some experts on these teams come from the NRC, while others come from industry. The NRC coordinates closely with the Institute of Nuclear Power Operations (INPO) in this process. In June 2013, the United States hosted a followup OSART mission to Seabrook Unit 1. The United States intends to continue to plan for an OSART mission every 3 years.

Since 1999, the NRC has participated in more than 25 IRRS mission teams, sending high-level technical experts on approximately four missions per year. In October 2010, the United States hosted an IRRS mission, focused on the U.S. operating reactor program. The NRC has devoted significant resources to addressing the mission's findings and implementing the team's recommendations. A followup mission is scheduled for early 2014. Additional information about the 2010 IRRS team's findings can be found in Section 8.1.5.2.

The NRC actively participates in the four IAEA Safety Standards Committees, the Nuclear Security Guidance Committee, and the Commission on Safety Standards. These activities, together with regular NRC staff participation in IAEA meetings to draft and revise safety standards and security guidance in coordination with other U.S. Government agencies, enable the NRC to use its broad regulatory experience to contribute to the safe and secure use of nuclear and radioactive materials in IAEA Member States.

The NRC also participates in the NEA Steering Committee, and holds leadership positions on NEA's Committee on the Safety of Nuclear Installations, the Committee on Nuclear Regulatory Activities, the Committee on Radiation Protection and Public Health, and the Radioactive Waste Management Committee. The NRC also holds leadership roles in, and is otherwise represented on, many of the NEA committee-chartered working groups. These activities provide diverse forums for nuclear regulators and research organizations to share information and work together to leverage resources for mutual benefit.

The NRC continues to participate in the Multinational Design Evaluation Program, with the goal of leveraging the experience of international counterparts in the review of new reactor designs. Through this program, the NRC is (1) sharing information with other regulatory authorities in the reviews of the Westinghouse's Advanced Passive (AP) 1000, Korea Hydro and Nuclear Power Company's APR1400, and AREVA Nuclear Power's U.S. Evolutionary Power Reactor (US EPR) designs, (2) cooperating in vendor inspections, and (3) pursuing possible convergence of regulations, codes, and standards associated with the design reviews of new reactors. In January 2013, the NRC Chairman assumed the chairmanship of the Multinational Design Evaluation Program.

Since the Fukushima Dai-ichi accident, the U.S. Government has augmented its coordinated program of international nuclear safety activities. Some existing activities have been expanded to address lessons learned from the accident, while certain initiatives were created specifically to address the accident and its implications. In both cases, the objectives of U.S. representatives in international meetings have been to expand their understanding of the accident and others' approaches to learning its lessons; to share relevant experience and lessons learned; and to minimize duplication of effort and leverage financial and human resources. In addition to contributing to the U.S. Government's direct cooperation with, and support of, the Government of Japan, the NRC has actively supported numerous post-Fukushima international activities, both on a bilateral and a multilateral basis. Through regular communication with its foreign Government counterparts, particularly in the regulatory area, and participation in international meetings, the United States has gained valuable information to enhance its domestic nuclear safety program and has contributed to the development of a stronger global nuclear safety regime. In particular, the NRC has worked closely with the IAEA in support of the Action Plan on Nuclear Safety, including participating in the various international experts' meetings, working through the IAEA Safety Standards Committees to address potential revisions to the safety standards, and providing consultants' advice on improving and enhancing the IAEA's suite of peer review services. In addition, the U.S. Government has supported at a high level both nuclear safety Ministerial Conferences, the first in Vienna in June 2011, and the second in Tokyo in December 2012. Appendix C provides more information on U.S. support for the IAEA's Action Plan on Nuclear Safety.

The NRC has also continued its work both with the IAEA and on a bilateral basis in support of countries seeking to develop new nuclear power programs or expand small or dormant programs. The NRC staff has been active in guidance document development in this area and has participated in numerous workshops and training activities to provide so-called "new entrant" countries with information and experience on building a robust, independent, regulatory infrastructure. To that end, the NRC has participated actively in the IAEA's Regulatory Cooperation Forum.

In addition to staff participation in more than 100 IAEA and NEA meetings each year, the NRC Chairman routinely participates in the IAEA General Conference and biannual meetings of the International Nuclear Regulators Association. Members of the Commission also travel to international conferences around the world to deliver keynote remarks, participate in panel discussions, and otherwise share insight on a variety of topics with diverse international audiences. The NRC's annual Regulatory Information Conference also provides a forum for international technical exchanges and high-level bilateral meetings, with more than 30 countries represented each year, many at senior levels.

<u>Bilateral Relations</u>. The NRC has arrangements to exchange technical information with nuclear safety agencies in more than 42 countries, Taiwan, and the European Atomic Energy Community. In addition, the NRC works with many other countries on a limited basis where there is not yet a formal bilateral arrangement in place. The NRC and its foreign counterparts routinely exchange operational safety data and other regulatory information. The NRC provides safety, security, emergency preparedness and safeguards advice, training, and other assistance to countries that seek U.S. help to improve their regulatory programs.

The NRC's information exchange arrangements serve as communication channels with foreign regulatory authorities, establishing a framework for the NRC to gain access to non-U.S. safety information that can (1) alert the U.S. Government and industry to potential safety problems, (2) help find possible accident precursors, and (3) provide accident and incident analyses, including lessons learned, which could be directly applicable to the safety of U.S. nuclear power plants and other facilities. The arrangements also serve as a vehicle for the assistance the NRC provides to countries to establish and improve their regulatory capabilities and infrastructure. Thus, the arrangements facilitate the NRC's strategic goal to support U.S. interests in the safe and secure use of nuclear materials and in nuclear nonproliferation.

Since the Fukushima accident, the NRC and its regulatory counterparts have shared a variety of information under the framework of these agreements, including preliminary results from the NRC's lessons learned activities. As the NRC's work in this area progresses and conclusions continue to develop, the NRC will continue to provide information about its activities and welcomes open, frequent exchanges of information to learn from its international counterparts. The NRC Chairman meets with foreign counterparts at the IAEA's annual General Conference. In addition, members of the Commission travel abroad to hold bilateral meetings with their regulatory counterparts, tour nuclear power plants and other facilities, and exchange information and good practices. Often, these visits result in increased communication between the NRC and its counterparts, providing opportunities for enhanced information exchange based on first-hand knowledge of various programs.

<u>International Assistance Programs.</u> In the early 1990s, the NRC began offering assistance to nuclear regulatory programs in several former Soviet states. The agency initially focused its efforts on those countries in which Soviet-designed reactors were operated. Following the September 11, 2001, terrorist attacks, the NRC expanded its assistance efforts to specifically include regulatory oversight assistance to countries that were considering or building new reactors, and assistance to improve regulatory oversight of radioactive sources. These efforts continue to expand under the International Regulatory Development Partnership, a collaborative program under the auspices of the NRC. The NRC provides technical assistance, training, and

generic documents covering a broad range of topics relevant to organizational infrastructure and regulatory programs relating to nuclear power programs. The NRC also maintains the Radiation Sources Regulatory Partnership, a program to assist countries with establishing and enhancing regulatory oversight of radioactive sources. Both of these programs are consistent with international legal commitments, such as the CNS and the Code of Conduct on the Safety and Security of Radioactive Sources, using IAEA standards as guidance. NRC consults with the IAEA and other member states to avoid duplication of activities.

Research Programs. The NRC conducts confirmatory regulatory research through the implementation of more than 100 bilateral and multilateral agreements in partnership with nuclear safety agencies and institutes in more than 30 countries. This research supports regulatory decisions on emerging technologies, aging equipment and facilities, and various other safety issues. The NRC and other nuclear regulatory and safety organizations carry out cooperative research projects to achieve mutual research needs with greater efficiency.

8.1.5.1 International Standards

The NRC actively participates in the development of the IAEA's safety standards. Where appropriate, the NRC also references the safety standards in NRC regulations.

The NRC is represented at the IAEA Commission on Safety Standards and all four IAEA Safety Standards Committees by senior executive managers. Over the course of a year, approximately 120 staff members support the multiple committee meetings. Additionally, the NRC provides senior expert assistance to the IAEA to support further development of the safety standards through the provision of cost-free experts, consultants, extrabudgetary support, and studies designed to advance the safety standards program.

The manner in which safety standards are used to inform and guide NRC regulations varies among the NRC's technical programs. For example, the IAEA's safety standards are used as reference documents to inform the development of requirements and guidance in the NRC's reactor, radiation protection, and waste management programs. Because of U.S. Government international legal commitments, the transportation safety documents are used directly in the U.S. transportation requirements. The United States made a political commitment to implement the IAEA Code of Conduct on the Safety and Security of Radioactive Sources. Subsequent to that policy decision, the Code of Conduct was embedded in the Energy Policy Act of 2005. As a result, the regulatory requirements in 10 CFR Part 110 were revised to incorporate the Code's guidance. Additionally, the Code of Conduct was used to inform the development of the U.S. National Source Tracking System.

Many of the differences in how the safety standards are applied to NRC regulations stem from the fact that NRC standards predate most IAEA safety standards. Further, the NRC requirements were written with a greater level of detail than the IAEA's safety standards. Despite these differences, the NRC agreed with recommendations from the 2010 IRRS mission to further harmonize requirements and guidance in the NRC's operating reactor program with IAEA safety standards. The NRC is actively working to implement these recommendations as NRC regulations and guides come up for periodic review. The NRC has revised its policy guidance and now directs staff to consider IAEA standards as a point of reference when drafting or revising regulatory documents, and to consider direct endorsement of the IAEA standards where appropriate. As a result of this directive, the NRC has published 23 new or revised regulatory guides that harmonize with or reference IAEA safety standards in the past 2 years.

The NRC anticipates increased attention to the IAEA safety standards and the IAEA Nuclear Security Series when revising NRC requirements across all programmatic areas.

8.1.5.2 Integrated Regulatory Review Service Mission

The NRC hosted an IAEA IRRS mission in October 2010, which focused specifically on the operating power reactor program. The mission report entitled "Integrated Regulatory Review Service (IRRS) Mission to the United States of America," was issued on March 1, 2011, and it contains 2 recommendations, 20 suggestions, and 25 good practices. The mission findings include that the NRC is a mature safety regulatory system that achieves its goals, has a transparent licensing process, and has a high level of human resource development. Suggestions include that the NRC should consider increasing its effort to use IAEA safety standards and should document the integration of its management system.

The NRC was pleased to host the IRRS mission to gain insights on, and identify improvements to, our operating power reactor program. NRC has invited a followup IRRS mission to occur in early 2014, to review both the NRC's response to the mission findings and its response to the Fukushima accident.

8.1.5.3 Operational Safety Assessment Review Teams

The NRC coordinates with INPO to implement the hosting of an OSART mission in the United States every 3 years. The United States welcomes the international views and knowledge exchanged through OSART, and to support and encourage this international program, the NRC credits the OSART mission in its Reactor Oversight Process by reducing some NRC inspection for the host nuclear power plant.

In June 2011, Seabrook Station Unit 1 hosted an OSART mission. The OSART team found several areas of good performance, including a healthy reporting culture. One of the identified areas for improvement is for the plant to be more proactive in resolving long-term issues. The Seabrook Station Unit 1 OSART report, "Final Report Operational Safety Review Team (OSART) Mission to Seabrook Nuclear Power Plant USA," was made public on March 16, 2012. The NRC reviewed the report and did not identify issues pertaining to either the plant or to the NRC's requirements that necessitate NRC program changes. A followup OSART mission was hosted in June 2013. Clinton Power Station, Unit 1, will host an OSART mission in 2014.

8.1.6 Financial and Human Resources

8.1.6.1 Financial Resources

As of September 30, 2012, the NRC had sufficient funds to meet program needs and adequate control of these funds in place to ensure it did not exceed budget authority. The sum of all funds available for fiscal year (FY) 2012 was $1,087.7 million, which is an $18.9 million decrease over the FY 2011 amount of $1,106.6 million

The NRC FY 2012 budget was financed with $909.5 million from user fees and $128.6 million from the U.S. Government's General Fund.

8.1.6.2 Human Resources

The NRC has developed a comprehensive human capital management system that is consistent with the agency's core values; reflective of its mission, strategic goals, and organization excellence objectives; clear in its purposes; and flexible in its implementation.

For the first time, both the Federal Employee Viewpoint Survey and the NRC Safety Culture and Climate Survey were administered in the same year. The U.S. Office of Personnel Management annually administers the Federal Employee Viewpoint Survey to Federal employees across all U.S. Government agencies, while the NRC's Office of the Inspector General Safety Culture and Climate Survey is an internal survey administered triennially to NRC employees only. Both surveys revealed that the NRC consistently scored above both Federal and private sector benchmarks, although in 2012 the agency did not perform as strongly as it had in the past. Governmentwide, NRC ranked #1 in the Knowledge Management and Talent Management indices; #2 in the Job Satisfaction index; and #3 in the Results-Oriented Performance Culture index. NRC also ranked #1 in the Global Satisfaction index, and #2 in the Employee Engagement index. The Federal Employee Viewpoint Survey showed significant gains in work-life programs, an area of focus from the last survey, as well as in communications, which was highlighted as a strength in the Safety Culture and Climate Survey. Opportunities for improvement exist in the areas of performance management and development, and the agency will focus action planning on these areas, along with improving the environment for raising concerns and valuing human differences.

Recruitment and Hiring Process. The NRC is focused on hiring the most critical skill sets, while still emphasizing governmentwide programs such as hiring of the disabled, employment of veterans, enhancing diversity, and supporting the agency's Comprehensive Diversity Management Plan. A number of internal and external factors are driving change at the NRC, including flat or decreasing agency budgets, lower than projected numbers of new reactors, and new work stemming from the Fukushima events. To meet current and future skill needs, the NRC is actively recruiting for its Nuclear Safety Professional Development Program, which has a history of graduating technically strong, diverse candidates. In addition, the NRC has maintained its recruitment activities at targeted universities and professional society conferences and career fairs. The agency advertises in trade journals and on Web sites to attract professionals in specialized technical disciplines.

The agency continued to make prudent, targeted use of recruitment, relocation, and retention incentives and pension offset waivers (rehiring annuitants without reduction of salary or pension) to hire and retain employees in mission critical positions. Such incentives are particularly useful for unusual occupations or highly specialized disciplines for which candidates may be scarce. The NRC continues to strengthen its programs for developing and hiring students in critical specialties through programs such as partnerships with colleges and universities that include university scholarship and fellowship grants, cooperative education programs, and payment of transportation and lodging expenses for student employees.

Retaining Staff. The NRC works to retain experienced staff, particularly those who are eligible to retire. In addition, the agency works to retain more recent recruits whose skills are highly marketable outside the agency. The NRC relies on all aspects of its human capital management system to retain staff. These include providing comprehensive training and development, constructive performance management, awards and recognition, opportunities for career growth, financial incentives when needed, and a range of benefits including health, wellness, and worklife programs. These worklife programs include flexible and alternative work schedules, as well as a robust flexiplace or telework program. The agency strives to create an Open, Collaborative Work Environment where people feel valued and challenged and where employees and leaders at all levels model the NRC's core values: integrity, service, openness, commitment, cooperation, excellence, and respect.

Training and Development. The NRC strives to maintain a learning culture where knowledge is shared throughout the organization. Such a culture supports the NRC's objective of sustaining a learning environment that provides continuing improvement in performance through knowledge management, performance feedback, training, coaching, and mentoring.

The NRC uses an integrated approach to learning to provide new employees with consistent information when it is needed. To assist new employees, the NRC has developed a virtual orientation center. This advanced training tool allows new hires to enter a computer-generated, or virtual, world where they can obtain information about the NRC organization, its mission, and employee benefits before starting their first day of work. Additionally, new hires receive position-specific training. The program offices have developed qualification programs that consist of three parts: general requirements, position-specific requirements, and oral qualification boards, for groups such as technical reviewers and project managers.

As an example, because of recent international events and a maturation of existing processes, there has been an increase in development of risk-informed licensing and regulatory applications. This has created a demand for PRA analysts at both the licensees and at the regulator. The NRC has responded to the challenge of development and retention of new analysts by initiating an in-house recruitment and training program. This Grow Your Own PRA Analyst Program is tailored to meet NRC staffing and training needs by building and maintaining a pool of qualified Reliability and Risk Analysts to address future risk assessment regulatory requirements. The program is designed to take internal candidates with diverse technical backgrounds, preferably with regulatory, nuclear power engineering, or operations backgrounds, and provide requisite training in various topics within PRA. The program is within its first year of application at the NRC.

The NRC continues to implement training technologies such as online and distance learning to deliver high quality learning products at a reduced cost. Over the past few years, the percentage of training conducted online has increased. In FY 2012, 82.2 percent of the 46,004 course completions were conducted as online courses as compared to 75.1 percent of the 39,307 course completions in FY 2011.

Leadership and Knowledge Management. The NRC has organized its leadership development programs into the Leaders Academy, consisting of competency-based training, assessment, and development programs for all levels of leadership, from individual contributors to senior executives. The NRC also continues its executive succession planning process, through which it identifies skills needed and potential successors for senior leadership positions, determines development that would benefit executives to prepare them for such NRC positions, and considers strategies for filling positions for which the NRC has few potential successors. This process informs selections for NRC positions and the establishment of executive development plans for all executives.

Knowledge management is a part of strategic human capital management, along with strategic workforce planning, recruitment, and training and development. As part of this effort, the NRC coordinates its activities to implement knowledge management strategies.

In addition, the NRC uses an agencywide knowledge management plan that serves as a framework to integrate new and existing approaches that generate, capture, and transfer knowledge and information relevant to the NRC's mission. This plan includes both near- and long-term strategies, such as the following:

- capture relevant critical knowledge of departing personnel
- recapture departed knowledge where possible
- communicate leadership expectations for a knowledge-sharing culture
- formalize knowledge management values and principles
- incorporate knowledge management within process workflows

Current knowledge management and knowledge transfer activities include the following:

- Branch Chief and Team Leader Seminars - As the role of the NRC branch chiefs has evolved from providing senior technical expertise to that of being a manager, it is essential that the branch chiefs have the information they need to succeed in their positions. As a community of practice, the branch chiefs and team leaders meet monthly to hear presentations by agency experts on topics such as performance management, budget, and communications.

- Branch and Team Meetings - To ensure that staff members in each branch or team are kept up-to-date in areas under their purview, branch chiefs and team leaders hold regularly scheduled staff meetings. During some of these meetings, senior staff members give presentations to staff regarding an area in which they are considered experts or to pass their knowledge of past events on to newer staff. Branch chiefs also provide opportunities for more junior staff to give presentations.

- Brown Bag Sessions – Informal meetings are held by individual offices to convey information or obtain feedback on common areas of interest, such as agencywide software system updates and upgrades, or changes to regulations and policies.

- NUREG/Knowledge Management (NUREG/KM) Series – NUREG/KM is a new series of NUREGs established to preserve knowledge of historical events that shaped the regulatory process. The first of the NUREG/KM series is an account of the Three Mile Island accident.

- Invitational Seminars and Panel Discussions – As part of knowledge transfer, former NRC employees are invited to share stories of their past experience on various relevant topics to give an historical perspective to NRC's current policies, processes, and procedures.

- Video Interviews – The NRC conducted a pilot project to capture knowledge from retiring senior staff using video interviews. One video captured knowledge about steam generators; another was entitled "Nuclear Knowledge for the Next Generation." The interviews included questions about licensing issues, recruiting and mentoring new hires, leadership, operations center experience, and reactor licensing performance metrics.

- Web sites – The NRC developed the "NRC Knowledge Center" site as an online venue for employees to collaborate, capture, and share knowledge to build organizational memory. The NRC Knowledge Center provides links to numerous topics, communities of practice, and informational videos to augment employees' learning and development. Office-specific knowledge management programs supplement this agencywide site. For example, the Office of Nuclear Reactor Regulation has a Web site devoted to knowledge management entitled “Sharing Expert Experience and Knowledge.” This site contains information such as the Inspector Best Practices Booklet and Inspector Newsletters, supervisor and team leader seminars, new employee orientation and training guide, key reference materials for reviews, qualification plans, strategic workforce planning, knowledge management, and other communities of practice.

8.1.7 Openness and Transparency

Openness has long been one of the NRC’s principles of good regulation as the agency seeks to carry out its mission of ensuring that radioactive materials are used safely and securely. The agency views nuclear regulation as the public’s business. As such, the NRC makes every effort to ensure its regulatory activities are open to the public.

Openness also requires the public to be able to participate meaningfully in the NRC’s regulatory processes. At the same time, the agency must control sensitive information so that it is not made public.

Openness is the second of six “Principles of Good Regulation” that the NRC first established in 1977. These principles guide all of the agency’s activities. Openness is also one of seven organizational values, adopted in 1995, to which the agency adheres in all its work. The NRC’s Strategic Plan, NUREG-1614, Volume 5, “Strategic Plan: Fiscal Years 2008-2013 (Updated),” issued February 2012, emphasizes Open Government principles and includes specific strategies for ensuring that the regulatory process, decisionmaking, and licensee oversight are all carried out as transparently as possible. That plan established five specific strategies to achieve openness, including the need to initiate early communication with stakeholders and to use clear and understandable language in communicating with the public.

Access to NRC Documents. From its inception, the NRC has made it a priority to maintain a Public Document Room, to assist the public in finding publicly available NRC information. The Public Document Room staff is comprised of skilled technical and reference librarians who provide information and research assistance directly to stakeholders, environmental groups, licensees, the legal community, and concerned citizens. The staff provides assistance in navigating the agency’s extensive collection of documents on licensing and rulemaking activities, as well as historical files from the NRC’s predecessor agency, the Atomic Energy Commission.

To ensure the public has access to the information it needs, the NRC makes all nonsensitive documents available to the public, unless there is a specific reason not to. The agency put in place policies, performance measures, and management controls to ensure this access. The NRC’s documents database, known as the Agencywide Documents Access and Management System (ADAMS), places all final records of publicly available documents into a searchable library that can be accessed through the NRC’s public Web site. The database includes documents and correspondence related to license applications, license renewals, and inspection findings. It does not include security-related, proprietary, or other sensitive information. During the final 8 months of 2012, the public accessed the ADAMS database more than 53,000 times and viewed more than 5.8 million documents.

The NRC measures and reports to Congress each year how quickly it releases internal and external documents, issues notices in advance of public meetings, and responds to requests filed under the Freedom of Information Act -- a Federal law giving the public the right to request and receive Government documents, with some exceptions.

The NRC also uses traditional tools to keep stakeholders informed. The agency sends copies of key documents and notifications to Federal, State, local, and Tribal authorities. The NRC also publishes notices in the *Federal Register* of Commission meetings, opportunities for hearings, and opportunities to comment on a variety of the agency's activities.

Open Government Plan.

The Open Government Plan designated one senior NRC manager as the accountability official, who, together with a senior advisory council, provides guidance on openness initiatives. The agency established a separate Open Government Advisory Group to oversee its Open Government program. These leaders work together to ensure that the agency has a continuing focus on adopting new technologies and making full use of their potential to reach out to and engage the public. The plan lays out the specific improvements the NRC is making to enhance stakeholder engagement.

The agency redesigned the NRC Web site to improve its search capabilities, introduced and made significant progress in social media, and published 29 high-value datasets—providing information that can improve public knowledge and agency accountability.

The NRC is an active participant in several governmentwide programs that promote transparency at the Federal level. These include www.data.gov, a Web site hosting high-value datasets; www.regulations.gov, an access portal for all Federal rulemakings; www.USAspending.gov, a Web site to which the NRC reports monthly all its spending on contracts, small purchases, and grants; www.itdashboard.gov, a site where the NRC and other agencies share details of their investments in information technology; and www.grants.gov, a source for finding and applying for Federal grants.

The NRC Web site. The NRC makes extensive use of its public Web site to share information with stakeholders and the public. During calendar year 2012, the NRC's Web site had more than 1.8 million individual visitors. The site was visited more than 4.1 million times, and visitors viewed more than 48 million pages. The site provides information on Commission decisions, hearing transcripts, inspection reports, enforcement actions, petitions, event reports, and daily plant status. It includes a tool that allows users to easily locate information on facilities the NRC regulates and details on the performance of reactor licensees. It also provides a great deal of general information and links to broaden the public's understanding of the NRC's mission, goals, and performance, as well as access to tools and information to help licensees and others to conduct business with the agency.

The site makes available all the NRC's press releases, issued when the agency receives license applications, makes major licensing decisions, takes enforcement actions, and announces public meetings, opportunities for hearings, and other avenues for public involvement. This information can also be provided automatically to anyone who requests a subscription. In fact, users may sign up through the Web site to automatically receive a number of different types of documents, including generic communications, new rulemaking dockets, speeches, and reports that the NRC's Inspector General issues. The public also can subscribe to receive correspondence related to specific facilities that interest them. The site includes an Open Government page with links to high-value datasets, information on the NRC's openness philosophy, and a tool allowing the public to suggest ways the agency can improve transparency, public participation, and collaboration.

The NRC video-streams high-interest Commission meetings over the Internet. More recently, the agency expanded Webcasting to other high-interest meetings, conferences, and adjudicatory hearings. These Webcasts are available for viewing live, as they occur, and are archived for viewing later.

The NRC has been working to identify additional Web-based tools that can expand its outreach. NRC staff received valuable input and suggestions for improving communications during a "virtual" public meeting held January 23, 2013. About 60 stakeholders called in and another 40 participated through a webinar. The staff is reviewing the input and has already taken action on some of the ideas participants shared.

Social Media Expands our Outreach. Over the past several years, the NRC has embraced social media as an important new tool for reaching a public audience beyond those with access to traditional communications tools. These social media platforms allow the agency to give information to the public, raise awareness, explain technical activities, and spotlight accomplishments. They also provide new vehicles for dialogue, giving the NRC new platforms to participate in two-way communication with the public. The NRC's Office of Public Affairs manages these tools, but NRC staff at all levels help to ensure the agency is meeting the communication needs of all our offices, both at headquarters and in the regions.

NRC's blog, or Web log, allows the agency to communicate with the public in plain language about topics of high interest or those that are complex from a technical or regulatory standpoint. The NRC posts blogs daily on a variety of subjects, and responds to comments posted by the public, as appropriate. The blog is a valuable tool for generating discussion about important matters. In the first 2 years, the agency posted about 300 blogs. Those posts have been viewed more than 300,000 times. The NRC received and posted more than 2,000 comments on the blog.

Social media platforms proved to be invaluable crisis communications tools during the events at Fukushima in 2011. Their value was again proven in October 2012, when several reactors shut down during Hurricane Sandy, a huge storm that affected the entire east coast of the United States. The most views the blog has ever had was on October 29, 2012, during the height of the storm. The blog was viewed more than 6,300 times that day. The blog has been instrumental in keeping the public informed on matters of great concern. Public comments submitted on the blog helped the NRC to develop content, as needed, to keep up with public demand for information.

The NRC's Twitter account offers an opportunity for the agency to quickly push out relevant information in a simplified format. For example, NRC can alert the public to new press releases, *Federal Register* notices, licensing decisions, guidance documents, important personnel changes, and any topic that might emerge. Eighteen months after launching the Twitter account, the NRC had more than 3,200 Twitter followers and continues to add new followers daily. The agency sent a total of 785 tweets during the first 18 months, for an average of 43 per month. During Hurricane Sandy, the NRC sent out 10 tweets, which were re-tweeted 130 times, potentially reaching more than 210,000 Twitter users.

The NRC has also created a YouTube channel and a Flickr photo gallery to provide a platform for video and image content and offer a gateway to additional information on the agency's Web site. The NRC posts photos and video of special events, important meetings, visits to nuclear facilities, and a variety of activities carried out by NRC staff. These forums enable the agency to document its work visually and introduce the people who carry out the agency's mission. In the first 18 months of using YouTube, the agency posted about 70 videos to the NRC YouTube channel. About 24,000 visitors viewed these videos and the NRC has about 230 subscribers who are notified each time new content is posted. In the first 12 months of using Flickr, the NRC posted more than 1,000 photos, viewed by about 50,000 visitors. The NRC's Flickr content has been viewed more than 104,000 times.

Public Meetings. The public continues to have many different opportunities to be involved in the NRC's regulatory decisionmaking process. Stakeholders may participate in a variety of ways before the agency issues certain licensing actions. To ensure this involvement is meaningful, the NRC actively communicates with stakeholders so they will understand how the NRC makes decisions – including the agency's role, processes, and activities. The NRC holds meetings with the public and other stakeholders near nuclear facilities, at agency headquarters, and at NRC regional offices.

The NRC is using a variety of tools to improve public participation. The agency is expanding its use of Web conferencing to allow participation by anyone with access to a computer, minimizing travel costs and increasing opportunities for public involvement. The agency actively seeks feedback from meeting participants to help identify ways the NRC can improve public meetings.

The NRC staff hosts and participates in a number of conferences, workshops, and symposia each year. The most prominent is the annual Regulatory Information Conference, which brings together more than 3,000 people from 30 countries, including members of Congress, nuclear industry representatives and other stakeholders. The conference features presentations by NRC's commissioners, NRC staff, licensees and other stakeholders. It serves as a communications vehicle to allow open dialogue on research findings, rulemakings, regulatory and safety issues, regulatory process and procedure improvements, international activities, and other items of interest. All presentations are available through the NRC Web site and the NRC Webstreams key events.

Details on the NRC's special programs for public involvement in oversight of operating nuclear facilities can be found in Section 6.3.11.

NRC's 2012 Open Government Plan describes goals for improving plain writing, high-value datasets, and services that the Public Document Room offers. The agency will continue efforts to strengthen social media services, expand the use of virtual meetings, and increase the visibility of rulemakings and NRC documents open for public comment. Improving the agency's use of plain language is an important goal for the immediate future.

The NRC has identified certain types of documents that should be written in plain language. They include informational brochures, performance assessments, generic communications, inspection reports, and significant enforcement actions. The agency is encouraging staff involved in preparing such documents to take plain language training, which the NRC offers both online and in a 2-day instructor led course.

The agency plans to strengthen the ability of stakeholders who use smart phones or other mobile devices to engage with the NRC. Under this initiative, the NRC will develop mobile-friendly Web pages and use quick response codes, enabling interested members of the public to scan barcodes for quick access to information.

8.2 Separation of Functions of the Regulatory Body from Those of Bodies Promoting Nuclear Energy

The partitioning of the U.S. Atomic Energy Commission in the mid-1970s resulted in distinct entities for the U.S. Government's regulatory and promotional responsibilities related to nuclear facilities and materials. Specifically, the Energy Reorganization Act redistributed the functions that the U.S. Atomic Energy Commission performed to two new agencies. This Act created the NRC to regulate the safety of the commercial nuclear power sector and the Energy Research and Development Administration (ERDA)—which has since become DOE—to promote nuclear power.

This Act gave the NRC independent authority to regulate the possession and use of nuclear materials as well as the siting, construction, and operation of nuclear facilities. The NRC performs its regulatory mission by issuing regulations, licensing commercial nuclear reactor construction and operation, licensing the possession of and use of nuclear materials and wastes, safeguarding nuclear materials and facilities from theft and radiological sabotage, inspecting nuclear facilities, and enforcing regulations. The NRC regulates the commercial nuclear fuel cycle materials and facilities. As to the regulatory control of commercial spent nuclear fuel and radioactive waste, the NRC is responsible for licensing commercial nuclear waste management facilities, independent spent fuel management facilities, and DOE facilities for the disposal of high-level radioactive waste and spent fuel.

ERDA addressed the U.S. Government's need to unify energy organization and planning. The DOE Organization Act brought a number of Federal agencies and programs, including ERDA, into a single agency with responsibilities for nuclear energy technology and nuclear weapons programs (i.e., DOE). Over the ensuing decades, DOE has expanded its new nuclear-related activities to include nonproliferation and the environmental cleanup of contaminated sites and facilities. With limited exceptions, DOE retains authority under the Atomic Energy Act for regulating its nuclear activities, including the responsibility for activities such as regulating the disposal of its own low-level radioactive waste.

8.3 Fukushima Lessons Learned

As a result of the Fukushima nuclear accident, there have been no changes in the U.S. legislative framework that governs the NRC and the regulations of the U.S. nuclear industry.

The U.S. Congress created the NRC as an independent regulatory agency in January 1975, with the passage of the Energy Reorganization Act. By giving the NRC an exclusively regulatory mandate, the statute reflected (in part) a decision by the U.S. Congress that the expanding commercial nuclear power industry, which was expected to continue to grow at that time, warranted the full-time attention of an exclusively regulatory agency. In creating the NRC, Congress also addressed a developing public concern that the Atomic Energy Commission's regulatory responsibilities were being overshadowed by the promotion of nuclear power. As discussed in Section 8.1, the NRC is headed by a five-member Commission. The President designates one member to serve as the Chairman and official spokesperson of the agency. The Commission as a whole formulates policies and regulations governing nuclear reactor and materials safety, issues orders to licensees, and adjudicates legal matters brought before it.

The NRC's place in the U.S. Government has not changed since Fukushima, though the NRC has had significant interaction with some entities in the Federal Government since the accident. The NRC has interactions with various entities in the Executive Branch, including the White House, OMB, U.S. Department of State, DOE, U.S. Environmental Protection Agency, FEMA, U.S. Department of Labor, U.S. Department of Transportation, U.S. Department of Defense, U.S. Department of Homeland Security, National Oceanic and Atmospheric Administration, and U.S. Department of Justice. After the Fukushima event, the NRC continued, and in some cases increased, communications with these entities.

Financial and human resource considerations related to Fukushima can be broken down into three categories:

(1) Incident response —the direct, immediate response to the Fukushima incident.
(2) Lessons learned development —the effort to define, document, and communicate lessons learned and a recommended path forward.
(3) Lessons learned implementation —the effort to request, review, and verify specific actions across licensees on the implementation of measures needed to gain alignment with the lessons learned.

Incident response to Fukushima required the following financial and human resource considerations:

- Staffing and maintenance of the headquarters Operations Center.
  - facilities maintenance costs on a 24-hour operating cycle (e.g., electricity, water, and air conditioning)
  - overtime paid to NRC staff
  - contract support
  - reduced capacity to support ongoing activities because senior management, mid-level management, and technical NRC staff were displaced from their normal duties
  - expanded operations center support staff
  - increased level of effort on the part of NRC staff not supporting Fukushima efforts to maintain existing operations while meeting existing operational goals
- Funding and NRC staff support for international travel.
- Overtime paid to Japan onsite technical staff support.

Fukushima lessons learned development supported by the efforts of the Near-Term Task Force (NTTF) and the Japan Lessons-Learned Project Directorate required the following financial and human resource considerations:

- Additional funding to support domestic and international travel.
- Temporary loss of experienced leadership and technical staff resources to support ongoing activities.
- An increased level of effort with respect to managing emergent requests for information regarding Fukushima-related activities.
- An increased level of effort to maintain existing operations while working towards existing operational goals.
- As a result of the accrual of credit hours and compensatory time earned in support of the increased workload, there is the potential for an increase in extended leave requests.

Lessons learned implementation is a limited multi-year ongoing effort with the following financial and human resource considerations:

- Funding for both domestic and international travel.
- Funding for contract support for review of licensee implementation plans.
- Reduced funding available to support other ongoing or planned activities.
- Budgeting for the Japan Lessons-Learned Project Directorate staff.
- Increased project management and technical support workload associated with Fukushima lessons learned licensee submittals.
- As a result of the accrual of credit hours and compensatory time earned in support of the increased workload, there is the potential for an increase in extended leave requests.

ARTICLE 9. RESPONSIBILITY OF THE LICENSE HOLDER

Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant license and shall take the appropriate steps to ensure that each such license holder meets its responsibility.

The NRC, through the Atomic Energy Act, ensures that the primary responsibility for the safety of a nuclear installation rests with the licensee. Steps that the NRC takes to ensure that each licensee meets its primary responsibility include the licensing process, discussed in Articles 18 and 19, the Reactor Oversight Process, discussed in Article 6, and the enforcement program, discussed below. This section provides an update on the licensee's responsibility for maintaining openness and transparency and a discussion on lessons learned from Fukushima.

9.1 Introduction

The NRC's regulatory programs continue to be based on the premise that the safety of commercial nuclear power reactor operations is the primary responsibility of NRC licensees. The agency is responsible for regulatory oversight of licensee activities to ensure that safety is maintained. The NRC reviews the safety of a reactor design and the capability of an applicant to design, construct, and operate a facility. If an applicant satisfies the Federal requirements, then the NRC will issue a license to operate the facility. Such licenses specify the terms and conditions of operation to which a licensee must conform. If a licensee does not conform to these license conditions, the NRC may take enforcement action, which can include modifying, suspending, or revoking the license. The NRC can also order particular corrective actions or issue civil penalties. The following sections discuss these enforcement mechanisms in greater detail.

9.2 The Licensee's Primary Responsibility for Safety

As discussed in Article 7 of this report, the Atomic Energy Act, Section 103, Chapter 10, grants the NRC authority to issue licenses for nuclear reactor facilities. Moreover, Section 103 states that these licenses are subject to such conditions as the NRC may establish by rule or regulation to implement the purposes and provisions of the Atomic Energy Act. Consistent with the Act, before issuing a license, the Commission determines that the applicant is (1) equipped and agrees to observe such safety standards to protect health and minimize danger to life or property as the Commission may establish by rule and (2) agrees to make available to the Commission such technical information and data about activities under such license as the Commission may determine necessary to promote the common defense and security and to protect public health and safety.

Embedded in each license is the explicit responsibility for the license holder to comply with the terms and conditions of the license and the applicable Commission rules and regulations. The licensee is ultimately responsible for the safety of its activities and the safeguarding of nuclear facilities and materials used in operation.

When the Commission determines that the licensee is not complying with the Commission's rules or regulations, the NRC takes appropriate action to ensure that the facility is returned to a condition compliant with its license. Details about the NRC's Enforcement Program are provided in the next section and in Section 7.2.4 of this report.

9.3 NRC Enforcement Program

As discussed in Article 7, the NRC has enforcement powers. As discussed in Sections 7.2.3 and 7.2.4, the enforcement process complements the Reactor Oversight Process. The NRC uses enforcement as a deterrent to emphasize the importance of compliance with regulatory requirements and to encourage prompt identification and prompt, comprehensive correction of violations.

The NRC identifies violations through inspections and investigations. All violations are subject to civil enforcement action and may be subject to criminal prosecution. Unlike the burden of proof standard for criminal actions (beyond a reasonable doubt), the NRC uses the Administrative Procedure Act standard (preponderance of evidence) in enforcement proceedings. After an apparent violation is identified, it is assessed in accordance with the Commission's enforcement policy, described in the NRC Enforcement Policy, last updated on January 28, 2013, which is available to NRC licensees and members of the public. The NRC Office of Enforcement maintains the current policy statement on the NRC's public Web site. Because it is a policy statement and not a regulation, the Commission may deviate from it, as appropriate, as the circumstances of a particular case may dictate.

The NRC has three primary enforcement sanctions available: notices of violation, civil penalties, and orders.[3] A notice of violation identifies a requirement and how it was violated; formalizes a violation pursuant to 10 CFR 2.201, "Notice of Violation"; requires corrective action; and normally requires a written response. A civil penalty is a monetary fine issued under authority of the Atomic Energy Act, Section 234, or the Energy Reorganization Act, Section 206. Section 234 of the Atomic Energy Act provides for penalties of up to $100,000 per violation per day; however, that amount is adjusted every 4 years by the Federal Civil Penalties Inflation Adjustment Act of 1990, as amended by the Debt Collection Improvement Act of 1996, and is currently $140,000. Section 161 of the Atomic Energy Act gives the Commission broad authority to issue orders; this authority extends to any area of licensed activity that affects public health and safety or the common defense and security. Orders modify, suspend, or revoke licenses, or they may require specific actions by licensees or persons. The NRC issues notices of violations and civil penalties on the basis of violations. The agency may issue orders for violations or, in the absence of a violation, because of a concern involving public health and safety or the common defense and security.

After identifying a violation, the NRC assesses its significance by considering the following factors:

- actual safety consequences
- potential safety consequences
- potential for impacting the NRC's ability to perform its regulatory function
- any willful aspects of the violation

3 The NRC also uses administrative actions, such as notices of deviation, notices of nonconformance, confirmatory action letters, and demands for information to supplement its enforcement program.

Given those factors, the NRC takes one of the following actions based on the significance of the violation:

- assigns a severity level, ranging from Severity Level IV (more than minor concern) to Severity Level I (the most significant)

- associates the violation with findings assessed through the Reactor Oversight Process significance determination process (described in Article 6) and assigns a color code of green, white, yellow, or red based on increasing risk significance

The Commission recognizes that there are violations of minor safety or environmental concern that are below Severity Level IV violations, as well as below violations associated with green findings. These minor violations are not assigned a severity level category or a color assessment.

The NRC may hold a predecisional enforcement conference or a regulatory conference with a licensee before making an enforcement decision if (1) escalated enforcement action appears warranted, (2) the NRC decides a conference is necessary, or (3) the licensee requests it. The purpose of the conference is to obtain information to assist the NRC in determining whether an enforcement action is necessary and, if so, what the appropriate enforcement action is. The conference focuses on areas such as (1) a common understanding of facts, root causes, and missed opportunities associated with the apparent violation and (2) a common understanding of the corrective actions taken or planned.

At several junctions during the enforcement process involving cases of discrimination or willful violation of NRC regulations, the agency offers its licensees (including their contractors) or individuals the opportunity to participate in the Alternative Dispute Resolution Program. Alternative dispute resolution is a general term encompassing various techniques for resolving conflict outside of court using a neutral third party. The NRC uses mediation, a technique in which a neutral mediator with no decisionmaking authority helps parties clarify issues, explore settlement options, and evaluate how best to advance their respective interests. Neutral mediators are selected from a roster of experienced mediators provided by a neutral program administrator who is under contract with the NRC. The mediator assists the parties in reaching an agreement. However, the mediator has no authority to impose a resolution upon the parties. Mediation is a confidential and voluntary process. If the parties to the process (the NRC and the licensee or individual) agree to use alternative dispute resolution, they select a mutually agreeable neutral mediator and share the cost of the mediator's services equally. In cases in which the NRC and the other party reach an agreement, the agency issues a confirmatory order reflecting the terms of the agreement.

The agency considers civil penalties for Severity Level I, II, and III violations, as well as knowing and conscious violations of the reporting requirements of Section 206 of the Energy Reorganization Act and the release of Safeguards Information by an individual. Although not normally used for violations associated with the Reactor Oversight Process, civil penalties (and the use of severity levels) are considered for issues that are willful, that have the potential to affect the regulatory process, or that have actual consequences.

Although each severity level may have several associated considerations, the outcome of the assessment process for each violation or problem (absent the exercise of discretion) results in one of three outcomes, which may involve no civil penalty, a base civil penalty, or twice the base civil penalty.

The NRC may issue orders to modify, suspend, or revoke a license; issue orders to cease and desist from a given practice or activity; or take other action as may be proper. The agency may issue orders in place of, or in addition to, civil penalties. Additionally, the NRC may issue an order to impose a civil penalty when a licensee refuses to pay a civil penalty or an order to an unlicensed person (including vendors) when the agency has identified deliberate misconduct. By statute, a licensee or individual may request a hearing upon receiving an order. Orders are normally effective after a licensee or individual has had an opportunity to request a hearing (i.e., 30 days). However, orders can be made immediately effective without prior opportunity for a hearing when the agency determines it is in the best interest of public health and safety to do so. Subsequent to the hearing process, a licensee or individual may appeal the administrative hearing decision to the Commission and, if desired, appeal the Commission's decision to a U.S. court of appeals.

Providing interested stakeholders with enforcement information is very important to the NRC. Conferences that are open to public observation appear in the listing of public meetings on the NRC's public Web site. The agency issues a press release for each proposed civil penalty or order. All orders are published in the *Federal Register*. Significant enforcement actions (including actions to individuals) are included in the enforcement document collection in the NRC's public Web site.

During 2010, the NRC issued a variety of significant enforcement actions to operating power reactors. These actions included 31 escalated notices of violation without civil penalties, 1 civil penalty, and 4 orders.

During 2011, the NRC issued a variety of significant enforcement actions to operating power reactors, including 30 escalated notices of violation without civil penalties, 0 civil penalties, and 1 order.

During 2012, the NRC issued a variety of significant enforcement actions to operating power reactors, including 40 escalated notices of violation without civil penalties, 2 civil penalties, and 10 orders.

To provide accurate and timely information to all interested stakeholders and enhance the public's understanding of the enforcement program, the NRC publishes related information on the agency's public Web site, and under certain circumstances, a press release.

9.4 <u>Openness and Transparency</u>

U.S. nuclear power plant licensees are required to demonstrate that the appropriate governmental authorities have the capability to alert the public of a nuclear power plant event (e.g., sirens, tone alert radios, and route alerting) and provide prompt, clear instructions on protective actions. At least annually, licensees provide members of the public located within the plume exposure pathway emergency planning zone information on how they would be notified and what their initial actions should be in an emergency as described in Section 16.8. Educational information on radiation, contact(s) for additional information, information on protective measures (e.g., evacuation routes and relocation centers, sheltering, respiratory protection, and radioprotective drugs), and direction to those needing assistance during an emergency is provided. A licensee's public information program includes the use of signs, notices, or other means, placed in areas such as motels, stores, and recreational venues for transient populations.

Each licensee has established a Joint Information Center that serves as a focal point for the coordination and dissemination of information from the licensee and Federal, State and local authorities to the public and media. In February 2011, the NRC published NUREG/CR-7032, “Developing an Emergency Risk Communication (ERC)/Joint Information Center (JIC) Plan for a Radiological Emergency,” and NUREG/CR-7033, “Guidance on Developing Effective Radiological Risk Communication Messages: Effective Message Mapping and Risk Communication with the Public in Nuclear Plant Emergency Planning Zones,” which address Joint Information Center enhancements to account for changes in media practices, advances in communications technology, and changes in public access to information and to address message mapping to support concise and consistent messaging.

The NRC’s openness and transparency objectives are described in Section 8.1.7 of this report.

9.5 Fukushima Lessons Learned

U.S. licensees continue to respond to NRC requests and initiatives that confirm and ensure adequate measures to protect public health and safety considering the lessons learned following the Fukushima nuclear accident.

In response to NRC Bulletin 2011-01, “Mitigating Strategies,” dated May 11, 2011, U.S. nuclear power plant licensees have provided comprehensive verification of their compliance with the regulatory requirements to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant because of explosions or fire.

U.S. licensees have openly engaged in NRC regulatory processes in response to Fukushima lessons learned, including the orders issued in March 2012, which require nuclear power plant operators to implement safety enhancements related to: (1) mitigation strategies to respond to external events resulting in the loss of power at plants, (2) ensuring reliable hardened containment vents for BWR Mark I and II designs, and (3) enhancing SFP instrumentation. The plant operators are required to begin implementation of the safety enhancements promptly and complete implementation within two refueling outages or by December 31, 2016, whichever comes first.

U.S. licensees have responded to RFIs issued by the NRC in March 2012, which requested each licensee to reevaluate the seismic and flooding hazards at its site using present-day methods and information, conduct inspections (or “walkdowns”) of existing seismic and flood protection features, and assess its emergency communications systems and staffing levels.

ARTICLE 10. PRIORITY TO SAFETY

Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear installations shall establish policies that give due priority to nuclear safety.

NRC policies that give due priority to safety covered under this article are PRA policy statements and policies that apply to licensee safety culture and safety culture at the NRC. Other articles (e.g., Articles 6, 14, 18, and 19) also discuss activities undertaken to achieve nuclear safety at nuclear installations. Updates to this section discuss lessons learned from Fukushima.

10.1 Background

The United States has made progress in developing and using the results of PRAs for all operating reactor facilities, and the NRC has developed extensive guidance on the role of PRA in U.S. regulatory programs. The agency has extensively applied information gained from PRA to complement other engineering analyses in improving issue-specific safety regulation and in changing the current licensing bases for individual plants. The move toward risk-informing the current regulations and processes continues to mark perhaps the most significant changes at the NRC. For example, regulations in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," modify the scope of the special treatment regulations by creating an alternative regulatory framework that enables licensees to use a risk-informed approach to categorize structures, systems, and components (SSCs), and their associated treatment, according to their safety significance. As another example, 10 CFR 50.48(c) allows an operating nuclear power plant licensee to adopt a risk-informed, performance-based fire protection program. The NRC is continuing a program to develop additional changes to the specific technical requirements in the body of 10 CFR Part 50.

10.2 Probabilistic Risk Assessment Policy

Three policy statements form the basis of the NRC's current treatment of PRA and the related regulatory safety goals and objectives - the "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985; the "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Republication," dated August 21, 1986; and the "Policy Statement on Use of PRA Methods in Nuclear Activities," dated August 16, 1995. Previous U.S. National Reports have detailed these policies.

10.3 Applications of Probabilistic Risk Assessment

The NRC applies PRA to resolve severe accident issues, evaluate new and existing requirements and programs, implement risk-informed regulation, and improve data and methods of risk analysis. The NRC also engages in cooperative activities with industry (such as pilot programs for 10 CFR 50.69 and 10 CFR 50.48(c)), and in activities that assess risk in determining plant-specific changes to the licensing basis. The NRC staff uses RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009, and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed License Amendment Requests after Initial Fuel Load," Revision 3, issued

September 2012, to assess the technical adequacy of the supporting PRA for all risk-informed applications.

The NRC maintains a risk-informed and performance-based plan, updated annually, which sets forth the agency's planned actions to make its regulatory activities risk informed and performance based. In the past, the Risk-Informed Regulation Implementation Plan (for example, SECY-09-0159 "Annual Update of the Risk-Informed and Performance-Based Plan" dated October 27, 2009), focused largely on risk-informed initiatives. The current improved plan has expanded the objectives to more fully achieve a risk-informed and performance-based regulatory structure. The NRC has created a public Web site for the risk-informed and performance-based plan with links to documents that specifically describe activities and status.

The NRC and industry representatives have cooperated in a number of activities and pilot programs to develop and apply risk-informed methodologies for specific regulatory applications. The staff uses the lessons learned from these activities to enhance the effectiveness of developed guidance. These activities, described in the sections below, include special treatment, inservice inspection, technical specification changes, and standards development.

For new reactors licensed under 10 CFR Part 52, the NRC requires applicants to describe the design-specific PRA and its results for a design certification application and a plant-specific PRA and its results for a combined license application. In addition, the NRC requires the holder of a combined license to develop a Level 1 and a Level 2 PRA before initial fuel load and it must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist. Each holder of a combined license must maintain and update the PRA every 4 years with upgraded consensus standards that exist at that time until operations permanently cease. Finally, before any application for license renewal, a combined license holder must upgrade the PRA to cover all modes and all initiating events.

10.3.1 Risk-Informed Special Treatment

The agency has approved a few applications of risk-informed inservice testing, of generally limited scope. In August 2001, the staff granted a risk-informed exemption request from the licensee of the South Texas Project regarding special treatment requirements for low-risk and non-risk-significant safety-related nuclear components (including an exemption from prescriptive inservice testing requirements). Having successfully implemented this exemption, the staff developed a new rule, 10 CFR 50.69, to allow the application of risk insights to reduce the special treatment requirements in 10 CFR Part 50 for SSCs that are categorized as being of low safety significance.

The Commission approved the final rule, with some modifications, in October 2004. The final rule was published in the *Federal Register* on November 22, 2004. The NRC staff issued RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Revision 1, in May 2006, for trial use.

A topical report, WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station," Revision 0, dated September 25, 2006, proposed a categorization process used by Wolf Creek Nuclear Operating Corporation in support of a future licensee submittal requesting approval to implement 10 CFR 50.69. The staff completed its review of the topical report and issued its final safety evaluation in March 2009. The staff found the categorization process described in the topical report to be acceptable, but it did not approve or endorse any specific treatment process. Treatment programs being implemented under 10 CFR 50.69 do not require prior approval from the NRC as part of the license amendment review process.

The staff has also developed guidance for sample inspections to be conducted at plants voluntarily choosing to implement 10 CFR 50.69. The performance of sample inspections is consistent with the statement of considerations accompanying the final 10 CFR 50.69 rule. The staff has issued draft guidance to obtain stakeholder input and has addressed those comments with the issuance of the final guidance. Inspection efforts will be focused on the most risk significant aspects related to implementation of 10 CFR 50.69 (i.e., proper categorization of SSCs and treatment of Risk-Informed Safety Class (RISC)-1 and RISC-2 SSCs). Additionally, the inspections are expected to be performance based, with SSCs with a lower safety significant function, such as those classified RISC-3, not receiving a major portion of inspection focus unless adverse performance trends are observed.

The staff recognizes the need for an effective, stable, and predictable regulatory climate for the implementation of 10 CFR 50.69. Inspection guidance developed with industry stakeholder input is viewed as an efficient vehicle for reaching a common understanding of what constitutes an acceptable treatment program for SSCs, since the NRC does not review specific treatment plans as part of a licensee's application to implement 10 CFR 50.69.

Recently, the NRC received an application for the Vogtle Nuclear Plant as a pilot application of 10 CFR 50.69. The lessons learned from this application review will be fed back into a future revision of the industry guidance and a revision to RG 1.201 and inspection guidance.

10.3.2 Risk-Informed Inservice Inspection

The NRC uses the guidance in RG 1.178, "An Approach for Plant Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," Revision 1, and NUREG-0800 Section 3.9.8, "Risk-Informed Inservice Inspection of Piping," both issued in September 2003, to evaluate applications of risk-informed inservice inspections. The agency has approved industry methodologies, one developed by the Westinghouse Owners Group and the other by EPRI, for alternatives to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Program.

ASME has also developed Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI Division 1." Code Case N-716 is founded, in large part, on the risk-informed inservice inspection process as described in EPRI Topical Report 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, issued December 1999, which the NRC reviewed and approved. Code Cases provide alternatives to existing ASME Code requirements that ASME has developed and approved. RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, issued October 2007, identifies the Code Cases that the NRC has determined to be acceptable alternatives to applicable parts of ASME Code, Section XI. RG 1.147 has not endorsed Code Case N-716 because the technical adequacy of a PRA that can be used to develop a risk-informed inservice inspection program was not well defined. The NRC has reviewed and approved about 12 plant-specific risk-informed inservice inspection programs that are based on the methodology described in Code Case N-716 supplemented with information related to the plant's PRA. By letter dated February 18, 2009, EPRI submitted for NRC staff review Topical Report 1018427, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs." The staff completed its review and endorsed Topical Report 1018427 with some comments. It is expected that a future revision of RG 1.147 will endorse Code Case N-716, supplemented by Topical Report 1018427. Licensees may implement Code Cases endorsed in RG 1.147 without prior NRC staff review and approval.

The NRC regularly participates in the ASME Code development process to resolve issues on risk-informed inservice inspection methodology.

10.3.3 Risk-Informed Technical Specification Changes

Since the mid-1980s, the NRC has reviewed and granted improvements to technical specifications that are based, at least in part, on PRA insights. In its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," published in the *Federal Register* on July 22, 1993, the Commission stated that it expects licensees to use a plant-specific PRA or risk survey in preparing submittals related to technical specifications. The Commission reiterated this point when it revised 10 CFR 50.36, "Technical Specifications," in July 1995.

The NRC continues to use RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued August 1998, and a companion section of NUREG-0800 to guide licensees in making risk-informed changes to plant technical specifications. The agency uses RG 1.177 as well as RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, issued May 2011, to improve plant technical specifications. The industry and the NRC continue to increase the use of PRA in developing improvements to technical specifications. As discussed in a letter from the Nuclear Energy Institute (NEI) to the NRC dated June 8, 2001 (Agencywide Documents Access and Management System Accession No. ML011690233), the industry developed eight separate initiatives to improve existing technical specification configuration control requirements through the use of risk insights. The following summarizes the major accomplishments in this area:

- Initiative 1, "Modified End States" - This initiative would allow (following a risk assessment) some equipment to be repaired during hot shutdown rather than cold shutdown. The NRC has approved the topical reports and model applications supporting this initiative for BWRs, Combustion Engineering, Babcock & Wilcox, and Westinghouse plants.

- Initiative 4b, "Risk-Informed Completion Times" - The overall objective of this initiative is to modify technical specifications to reflect a configuration risk management approach that is more consistent with the approach of the Maintenance Rule (10 CFR 50.65(a)(4)). Industry guidance has been approved, and the South Texas Project pilot was approved in 2007. The NRC has approved a model application for this initiative. The NRC received a combined license application for Comanche Peak, Units 3 and 4, which also included Initiative 4b. The application is currently under review.

- Initiative 5b, "Risk-Informed Method for Control of Surveillance Frequencies" - This initiative allows licensees to modify the frequency of technical specification surveillances based on test data and a risk-informed evaluation. The staff approved industry guidance and a model application, and it has approved pilot applications for the Limerick Generating Station in 2006, and Diablo Canyon in 2009. The staff continues to receive and review applications for this initiative. Initiative 5b is also included in a combined license application for Comanche Peak, Units 3 and 4.

- Initiative 6, “Modification of Limiting Condition for Operation 3.0.3, Actions and Completion Times” - This initiative provides a 24-hour completion time for a limited scope of technical specification systems when both safety trains are inoperable. The NRC has approved the Combustion Engineering Topical Report WCAP-16125, “Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,” Revision 2, issued May 2009. The NRC is in the process of reviewing a model application for Combustion Engineering plants.

- Initiative 7, “Non-Technical Specifications Support System Impact in Technical Specifications System Operability” - This initiative permits a risk-informed delay time before entering limiting condition for operation actions for inoperability attributable to a loss of support function provided by equipment not addressed in technical specifications. Guidance documents have been approved for snubbers and hazard barriers. The staff continues to receive and review applications for this initiative.

- Initiative 8, “Remove/Relocate Non-Safety and Non-Risk Significant Systems from Technical Specifications” - This initiative would review technical specifications to remove certain system functions that had been included solely because they were judged to be risk significant at one time, but additional analysis could show them not to be. The industry and staff are in preliminary discussions on this initiative.

10.3.4 Development of Standards

The NRC worked with ASME and the American Nuclear Society (ANS) to develop a national consensus standard for PRA quality. In February 2009, ASME and ANS issued their joint PRA quality standard, ASME/ANS-RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” and the NRC endorsed it in RG 1.200, Revision 2, in March 2009. This PRA quality standard addresses all hazards at full power operations for core damage frequency (Level 1 PRA) and large early release frequency (aspect of Level 2 PRA) for current light water reactor designs.

The agency plans further revisions to the RGs to incorporate revisions to the ASME/ANS standards as they are published, including standards addressing low power and shutdown modes, Level 2 and 3 PRA, and advanced light water and non-light water reactor designs.

10.4 <u>Safety Culture</u>

An important way to implement any policy that gives due priority to safety is to foster a strong safety culture in the organization. The following discussion focuses on safety culture, and NRC efforts to improve safety culture.

The NRC issued its “Final Safety Culture Policy Statement” in the *Federal Register* in June 24, 2011. This policy statement outlines the Commission’s expectation that all licensees maintain a positive safety culture at their facilities. The NRC defines nuclear safety culture as the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment. This policy statement applies to all licensees, certificate holders, permit holders, authorization holders, holders of quality assurance program approvals, vendors and suppliers of safety-related components, and applicants for a license, certificate, permit, authorization, or quality assurance program approval, subject to NRC authority.

The NRC has identified the following traits of a positive safety culture:

- Leadership safety values and actions - leaders demonstrate a commitment to safety in their decisions and behaviors;
- Problem identification and resolution - issues potentially affecting safety are promptly identified, fully evaluated, and promptly addressed and corrected commensurate with their significance;
- Personal accountability - all individuals take personal responsibility for safety;
- Work processes - the process of planning and controlling work activities is implemented so that safety is maintained;
- Continuous learning - opportunities to learn about ways to ensure safety are sought out and implemented;
- Environment for raising concerns - a safety conscious work environment is maintained in which personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment, or discrimination;
- Effective safety communication - communications maintain a focus on safety; and
- Respectful work environment - trust and respect permeate the organization.

10.4.1 NRC Monitoring of Licensee Safety Culture

This section covers the policies, programs, and practices that apply to licensee safety culture.

10.4.1.1 Background

Section 6.3.2 of this report describes the Reactor Oversight Process. Based on lessons learned from the Davis-Besse reactor pressure vessel head degradation event and other considerations, the NRC enhanced the Reactor Oversight Process to more fully address safety culture and identify safety culture problems earlier so that corrective steps can be taken to address the problems and prevent further plant performance degradation.

10.4.1.2 Enhanced Reactor Oversight Process

The NRC currently uses the IAEA International Nuclear Safety Advisory Group's definition of safety culture provided in Safety Series No.75-INSAG-4, “Safety Culture,” issued February 1991, as “that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear safety issues receive the attention warranted by their significance.”

On the basis of a review of safety culture attributes developed or applied by IAEA, NEA, INPO, regulatory bodies in other countries, and other domestic organizations, staff expertise, and input and feedback from NRC stakeholders, the staff identified the following components as important to safety culture:

- decisionmaking
- resources
- work control
- work practices

- corrective action program
- operating experience
- self- and independent assessments
- environment for raising safety concerns
- preventing, detecting, and mitigating perceptions of retaliation
- accountability
- continuous learning environment
- organizational change management
- safety policies

The Reactor Oversight Process inspection guidance documents define each one of the safety culture components in a greater level of detail (e.g., cross-cutting aspects). The Reactor Oversight Process applies the safety culture components, and their associated aspects, in different ways. The first nine safety culture components are applied in the baseline inspection and assessment program. All 13 safety culture components are applied in selected baseline, event followup, and supplemental IPs.

After publication of the safety culture policy statement, the NRC engaged INPO, NEI, and external stakeholders to develop a common safety culture language. This language, which was finalized in early 2013, better aligns the industry's language with the NRC's language to allow for more clarification and enhance understanding of licensee performance. The NRC plans to update all guidance and inspection documents appropriately with the new common safety culture language in 2013.

Licensees perform periodic, voluntary self-assessments of safety culture in accordance with industry guidelines. There are no regulatory requirements for licensees to perform safety culture assessments routinely. However, depending on the extent of deterioration of licensee performance, the NRC has a range of expectations about regulatory actions and licensee safety culture assessments, as described below.

The Reactor Oversight Process uses a graded approach, such that plants that are performing in a specified manner warrant only a routine level of inspection and oversight. However, as licensee performance deteriorates, inspection and oversight become increasingly more intrusive to ensure safe plant operation. The Reactor Oversight Process safety culture enhancements continue to allow licensees to self-diagnose and implement corrective actions for their performance problems before the NRC performs followup inspections.

For most licensees (i.e., those listed in the Licensee Response column, Column 1, of the Reactor Oversight Process Action Matrix), the NRC performs the baseline inspection program. In the routine or baseline inspection program, the inspector will develop an inspection finding and then identify whether an aspect of a safety culture component is a significant causal factor of the finding. The NRC communicates the inspection findings to the licensee along with the associated safety culture aspect.

When performing the IP that focuses on problem identification and resolution, inspectors have the option to review licensee self-assessments of safety culture. The problem identification and resolution IP also instructs inspectors to be aware of safety culture components when selecting samples. In addition, the procedure contains enhanced questions related to a safety-conscious work environment.

IP 71153, “Followup of Events and Notices of Enforcement Discretion,” dated December 5, 2011, directs inspection teams to consider contributing causes related to the safety culture components as part of their efforts to fully understand the circumstances surrounding an event and its probable cause(s).

As part of the assessment process (conducted semi-annually), the NRC considers the aspects of safety culture components associated with inspection findings to determine whether common themes exist at a plant. If, over three consecutive assessment periods (i.e., 18 months), a licensee has the same safety culture issue with the same common theme, the NRC may ask the licensee to conduct a safety culture self-assessment.

As licensee performance declines (Regulatory Response column, Column 2, of the Reactor Oversight Process Action Matrix), the inspectors, through a specific supplemental IP, verify that the licensee’s root cause, extent of condition, and extent of cause evaluations for the risk-significant finding(s) appropriately considered the safety culture components.

If the licensee performance degrades further (Degraded Cornerstone column, Column 3, of the Reactor Oversight Process Action Matrix), the NRC expects that the licensee’s root cause evaluation for the risk-significant finding(s) will determine whether any safety culture component contributed to the risk-significant performance issues. If, through the conduct of supplemental IP 95002, “Inspection for One Degraded Cornerstone or any Three White Inputs in a Strategic Performance Area”, dated February 11, 2011, the NRC determines that the licensee did not recognize that safety culture components caused or significantly contributed to the risk-significant performance issues, the NRC may request the licensee to complete an independent assessment of its safety culture.

Finally, for licensees with more significant performance degradation (Multiple/Degraded Cornerstone column, Column 4, of the Reactor Oversight Process Action Matrix), the NRC will expect the licensee to conduct a third-party independent assessment of its safety culture. The NRC will review the licensee’s assessment and will conduct an independent assessment of the licensee’s safety culture through a specific supplemental IP that was substantially revised to provide guidance for these assessments.

Considerations of safety culture within the Reactor Oversight Process provide the NRC staff with (1) better opportunities to consider safety culture weaknesses and to encourage licensees to take appropriate actions before significant performance degradation occurs, (2) a process to determine the need to specifically evaluate a licensee’s safety culture after performance problems have resulted in the placement of a licensee in the Degraded Cornerstone column of the Reactor Oversight Process Action Matrix, and (3) a structured process to evaluate the licensee’s safety culture assessment and to independently conduct a safety culture assessment for a licensee in the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

By using the existing Reactor Oversight Process framework, the NRC’s safety culture oversight activities are based on a graded approach and remain transparent, understandable, objective, risk-informed, performance-based, and predictable.

10.4.2 The NRC Safety Culture

Given the NRC's safety and security mission, the NRC recognizes the importance of maintaining its own strong safety culture and the need to continuously seek to improve its internal organizational effectiveness.

In response to the identification of licensee safety culture weaknesses as contributing factors to events, the agency revised the Reactor Oversight Process in 2006, to better address safety culture; enhancement efforts to the Reactor Oversight Process continue. These external efforts prompted internal reflection on how to improve the agency's own safety culture. Accordingly, in October 2008, the agency chartered the NRC Internal Safety Culture Task Force to provide a report to the Commission outlining potential initiatives that could improve the agency's internal safety culture.

Based on the results from a range of data collection activities and the experience and knowledge of its members, the NRC Internal Safety Culture Task Force developed a set of recommendations. These recommendations, which are being implemented, aim to create effective and lasting improvements for supporting a strong safety culture. Actions include the following:

- appointing an agency Safety Culture Program Manager
- integrating safety culture into the NRC's Strategic Plan and integrating performance management tools
- developing training on safety culture principles and expectations
- evaluating the agency's problem identification, evaluation, and resolution processes
- establishing clear expectations and accountability for maintaining current policies and procedures

SECY-09-0068, "Report of the Task Force on Internal Safety Culture," dated April 27, 2009, and SECY-10-0009, "Internal Safety Culture Update," dated January 26, 2010, provide more details, including, in the latter, a status on the implementation of the recommendations in the task force report.

Complementing this new initiative is the agency's ongoing effort to encourage the free and open discussion of differing professional views to develop sound regulatory policy and decisions. The NRC strives to establish and maintain an Open, Collaborative Work Environment that encourages all employees and contractors to voice differing views promptly without fear of retaliation. In 2007, the NRC created an Open, Collaborative Work Environment internal Web page for the staff, which clearly communicates that the NRC encourages trust, respect, and open communication to foster and promote a positive work environment that maximizes the potential of all individuals and improves regulatory decisionmaking. The Web page also identifies some of the policies in place that permit employees at all levels in all areas to provide professional views on virtually all matters pertaining to the agency's mission.

The NRC Open Door Policy (first communicated to agency employees in 1976), the NRC Differing Professional Opinions Program (formally established in 1980), and the NRC Non-Concurrence Process (established in 2006) illustrate the NRC's commitment to the free and open discussion of professional views. In 2008, the NRC created the NRC Team Player Awards, which recognize and celebrate behaviors that support an Open, Collaborative Work Environment where differing views are welcomed, valued, fairly considered, and addressed.

The agency uses the Office of the Inspector General's periodic Safety Culture and Climate Survey to assess the effectiveness of these new and existing safety culture efforts. In 1998, the Office of the Inspector General conducted the first in a continuing series of Safety Culture and Climate Surveys to identify areas for additional organizational improvements. The surveys are voluntary, provide for anonymity, and are offered to all NRC employees, supervisors, and managers. In addition, a survey makes it possible to compare category-level results for the NRC to other U.S. organizations that have completed such a survey. The Office of the Inspector General has conducted the Safety Culture and Climate Surveys five times: 1998, 2002, 2005, 2009, and most recently in 2012.

The NRC is addressing the survey responses to maintain areas identified as strengths and to improve areas identified as challenges. The staff is developing office and agency action plans and conducting agencywide focus groups to gain further insight into survey findings to pursue continuous improvement in both safety culture and organizational effectiveness.

10.5 Managing the Safety and Security Interface

Safety and security have always been the primary pillars of the NRC's regulatory programs. Safety and security activities have become closely intertwined, and it is critical that consideration of these activities be integrated so as not to diminish or adversely affect either safety or security. While many safety and security activities complement each other, there is the potential for security measures to inadvertently affect plant safety, or for safety activities to inadvertently affect security. Recognizing this potential for adverse impact, the NRC has increased its attention to the interfaces between these two areas.

The NRC's mission statement and strategic goals establish a firm foundation for its regulatory framework that stresses the importance of maintaining both safety and security. The NRC is implementing many efforts in the areas of rulemaking, licensing and inspection to recognize, establish, and improve this interface. The NRC has been working multilaterally with the IAEA and bilaterally with its international counterparts to promote this concept. In March 2009, the NRC issued 10 CFR 73.58, "Safety/Security Interface Requirements for Nuclear Power Reactors," which requires licensees to assess and manage changes to safety and security activities so as to prevent or mitigate potential adverse effects that could negatively impact plant safety or security. In addition, as part of the reactor security rulemaking effort, the NRC developed guidance on safety and security interfaces at nuclear power plants in RG 5.74, "Managing the Safety/Security Interface," issued June 2009.

From 2004 to 2012, as part of the NRC's increased focus on security events following the events of September 11, 2001, security issues were considered in the Reactor Oversight Process through a different assessment process than safety issues. On July 20, 2011, the Commission issued SRM-SECY-11-0073, "Staff Proposal to Reintegrate Security into the Action Matrix of the Reactor Oversight Process Assessment Program," approving the reintegration of the security cornerstone in the reactor assessment process. As described in Regulatory Issue Summary (RIS) 2012-03, "Reintegration of Security into the Reactor Oversight Process Assessment Program," dated March 14, 2012, this reintegration became effective on July 1, 2012. This reestablished the original framework of the Reactor Oversight Process, which involved a holistic assessment of licensee performance.

Satisfactory licensee performance in the Reactor Oversight Process cornerstones provides reasonable assurance of safe and secure facility operation and that the NRC's safety and security missions are being accomplished. Like the other cornerstones, the security cornerstone contains IPs and performance indicators to ensure that its objectives are being met. NRC addresses the safety and security interface issues in evaluating their implications among the cornerstones and in the cross-cutting areas of human performance, safety conscious work environment, and problem identification and resolution. Safety and security are integrated into the NRC's regulatory framework and evaluated by the NRC staff using an integrated assessment process. To ensure licensees are complying with the regulations, the NRC has incorporated the evaluation of the licensee's interfaces with nuclear security into its IPs.

The section of this report on nuclear programs and Section 6.3.2 of this report discuss the Reactor Oversight Process in more detail.

The NRC also recognizes the impact that organizational safety culture has on both safety and security, as well as on the interface between the two areas. The ongoing effort to implement the NRC's Safety Culture Policy Statement is described in detail in Section 10.4 of this report.

10.6 <u>Fukushima Lessons Learned</u>

The last NRC-sponsored Level 3 PRA was performed more than 20 years ago. The NRC staff is developing plans for a new Level 3 PRA. This decision was driven primarily by technical advances, which include: (1) plant modifications to enhance nuclear power plant operational performance, safety, and security; (2) improved understanding and modeling of severe accident phenomena; and (3) advances in PRA technology, such as common-cause modeling.

In response to the Fukushima lessons learned, the NRC staff has identified additional scope considerations that could be addressed in a new and more comprehensive Level 3 PRA. These factors include: (1) multi-unit site effects; (2) other site radiological sources (e.g., SFPs or dry storage casks); and (3) site-specific external hazards such as fires, flooding, and seismic events.

A new full-scope, comprehensive, site Level 3 PRA that incorporates these technical advances and additional scope considerations could improve the NRC's understanding of probable risk, enhancing regulatory decisionmaking and helping the agency focus its limited resources on issues most pertinent to its mission to protect public health and safety. On September 21, 2011, the Commission directed that a full-scope comprehensive site Level 3 PRA for an operating plant be completed within 4 years. The new Level 3 PRA will offer insight into many of the NTTF recommendations.

ARTICLE 11. FINANCIAL AND HUMAN RESOURCES

1. Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation throughout its life.

2. Each Contracting Party shall take the appropriate steps to ensure that sufficient numbers of qualified staff with appropriate education, training, and retraining are available for all safety-related activities in or for each nuclear installation, throughout its life.

This section explains the requirements about financial resources that licensees must have to support the nuclear installation throughout its life, and the regulatory requirements for qualifying, training, and retraining personnel. It also discusses lessons learned from Fukushima.

11.1 Financial Resources

Reasonable assurance of adequate funds for the safe construction, operation, and decommissioning of nuclear installations is necessary for the mission to protect public health and safety. Although there does not appear to be a consistent relationship between a licensee's finances and operational safety, some evidence suggests that financial pressures may limit the resources devoted to corrective actions, plant improvements, and other safety-related expenditures. Furthermore, because a power reactor must operate to supply the revenues for eventual plant decommissioning, any shutdown of a plant before its owner has accumulated sufficient funds for decommissioning could potentially hinder the safe decommissioning of that plant.

Additionally, many U.S. States have undertaken economic deregulation of nuclear power plants. Traditionally, nuclear power plant owners have been large, vertically integrated companies with substantial assets in generation, transmission, and distribution. In exchange for having exclusive franchises to supply electric power in defined geographical areas, nuclear power plant owners have had the rates they charge to their customers regulated by State Government agencies. This system of rate-based regulation has ensured a source of funds for construction, operation, and decommissioning of nuclear power plants. Nonetheless, this model of rate-based regulation has been changing and the NRC has adjusted its processes in response.

The NRC distinguishes among financial qualifications for construction, operation, and decommissioning of nuclear power plants, and has separate regulations and programs that apply to each. The NRC also implements programs to ensure that the public has financial protection for bodily injury and property damage losses in the event of a nuclear incident. Finally, the agency has implemented requirements to ensure that licensees have insurance to help pay onsite recovery costs resulting from accidents and to supply funds for postaccident restart or decommissioning.

11.1.1 Financial Qualifications Program for Construction and Operations

This section explains the financial qualifications program for construction and operations and describes NRC reviews for construction permits, operating licenses, combined licenses, postoperating nontransferred licenses, and license transfers.

Section 182.a of the Atomic Energy Act provides that "each application for a license … shall specifically state such information as the Commission, by rule or regulation, may determine to be necessary to decide such of the technical and financial qualifications of the applicant … as the Commission may deem appropriate for the license." To implement this provision, the NRC has developed the regulations and guidance discussed below.

11.1.1.1 Construction Permit Reviews

As required by 10 CFR 50.33(f)(1), applicants for construction permits must submit information that "demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs." Appendix C, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits," to 10 CFR Part 50 gives more specific directions for evaluating the financial qualifications of applicants.

11.1.1.2 Operating License Reviews

An "electric utility" as defined in 10 CFR 50.2, "Definitions," is "any entity that generates or distributes electricity and which recovers the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority." Electric utilities are exempt under 10 CFR 50.33(f) from reviews of financial qualifications of applications for operating licenses. The reason for this exemption is that cost-of-service rate regulation, as it has existed in the United States, has ensured that ratepayers provide a source of funds for the safe operation of nuclear power plants. Applicants for operating licenses that are not electric utilities are required under 10 CFR 50.33(f)(2) to submit information that demonstrates that they possess or have reasonable assurance of obtaining the necessary funds to cover estimated operating costs. Nonelectric-utility applicants for operating licenses also are required to submit estimates for the total annual operating costs for each of the first 5 years of operation of their facilities and must state the sources of funds to cover operating costs.

11.1.1.3 Combined License Application Reviews

As authorized in 10 CFR Part 52, applicants may apply for a combined construction permit and operating license. Under 10 CFR 52.77, "Contents of Applications; General Information," such applications must contain all of the information required under 10 CFR 50.33, "Contents of Applications; General Information," including information about financial qualifications. The NRC uses the procedures described above to review future combined license applications.

11.1.1.4 Postoperating License Nontransfer Reviews

The NRC does not systematically review the financial qualifications of power reactor licensees once it has issued an operating license, other than for license transfers as described below. However, as provided in 10 CFR 50.33(f)(5), the NRC can seek additional information on licensees' financial resources if the agency considers such information appropriate. For example, the staff may review financial and industry trade press as well as other publicly available information, such as Securities and Exchange Commission (SEC) submissions and Federal Energy Regulatory Commission (FERC) submissions to identify potential changes in licensees' financial health. If the review of any of these sources indicates that a licensee's financial health may be deteriorating, the NRC can request additional financial information from the licensee as authorized by 10 CFR 50.33(f)(5) to confirm that a licensee has the financial resources to operate the facility safely.

11.1.1.5 Reviews of License Transfers

The NRC regulations in 10 CFR 50.80, "Transfer of Licenses," require agency review and approval of transfers of operating licenses, including licenses for nuclear power plants owned or operated by electric utilities. The NRC performs these reviews to determine whether a proposed transferee or new owner is technically and financially qualified to hold the license.

NUREG-1577, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1, issued February 1999, describes the agency's overall review process of applicant and licensees' financial qualifications for nuclear power plant construction and operation.

11.1.2 Financial Qualifications Program for Decommissioning

Among other sections of the Atomic Energy Act, Section 182.a establishes the basis for the NRC's regulations and guidance on decommissioning funding assurance. In addition, 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," gives the requirements for licensee recordkeeping and reporting of nuclear decommissioning funds to the NRC.

11.1.3 Financial Protection Program for Liability Claims Arising from Accidents

The Price-Anderson Act of 1957, which became Section 170 of the Atomic Energy Act, governs the U.S. financial protection program. Along with related definitions in Section 11, Section 170 supplies the financial and legal frameworks to compensate those who suffer bodily injury or property damage as a result of accidents at nuclear facilities covered by the law. The NRC regulations implementing the provisions of Section 170 for NRC licensees are codified in 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements."

The Price-Anderson Act was enacted to (1) remove the deterrent to private-sector participation in atomic energy presented by the threat of potentially enormous liability claims in the event of a catastrophic nuclear accident and (2) ensure that adequate funds are available to the public to satisfy liability claims if such an accident were to occur.

The Price-Anderson Act was revised most recently in 2005, when Congress renewed the Commission's authority to cover new facilities until 2020. Under the current law, power reactors over 100 megawatts electric must contribute to a funding pool that replaces the U.S. Government as the second provider of funds if the first layer of financial protection (liability insurance, now $375 million) is exhausted. The NRC is required by Section 170t to adjust these amounts for inflation every five years based on the aggregate change in consumer price index.

After an accident, reactor operators must pay into a "retrospective premium pool" in maximum annual installments not to exceed $15 million, up to a total of $111.9 million each. But payment is called for only if the accident exhausts the first layer of financial protection, and only if and to the extent that, additional funds are needed to pay the damages. With 104 reactors currently participating in the system, the total financial protection available under the Price-Anderson Act for any one accident is approximately $12 billion ($375 million primary coverage plus $111.9 million per reactor times 104 reactors), which is also the limit on liability. As reactors leave the retrospective premium system as a result of permanent closure or join as the result of construction of new reactors, this coverage limit may fall or rise. A change in the limit also may occur when the $111.9 million contribution is adjusted for inflation, as must be done every 5 years. In any event, Congress will address any damages exceeding the total sum that reactors must contribute to the pool and will decide upon the next steps needed for compensation.

The public benefits significantly from another feature of the Price-Anderson Act. Claimants need only prove that the accident caused their injury to receive compensation for damages from any accident with significant offsite releases of radiation (i.e., an "extraordinary nuclear occurrence"). Neither proof of fault nor proof of what caused the accident is necessary.

Claims for more than 150 alleged incidents involving nuclear material have been filed under various liability policies since the inception of the Price-Anderson Act in 1957. The insured losses and expenses paid so far total more than $125 million. Most payments arose out of the accident at Three Mile Island Unit 2.

11.1.4 Insurance Program for Onsite Property Damages Arising from Accidents

Among other sections of the Atomic Energy Act, Section 182.a gives the basis for the NRC's onsite property damage insurance requirements for operating nuclear power reactors contained in 10 CFR 50.54(w).

The U.S. nuclear industry has not experienced an accident involving radioactive release since the Three Mile Island Unit 2 event in 1979.

11.2 <u>Regulatory Requirements for Qualifying, Training, and Retraining Personnel</u>

This section explains the regulatory requirements for qualifying, training, and retraining personnel. It discusses the governing documents, the process for implementing requirements, and experience. It also discusses INPO accreditation activities.

11.2.1 Governing Documents and Process

The NRC regulates the training requirements for licensed operators and licensed senior operators under 10 CFR Part 55, "Operators' Licenses," which allows facility licensees to have operator requalification program content that is derived using a systems approach to training (SAT), as defined in 10 CFR 55.4, "Definitions," or that meets the requirements outlined in 10 CFR 55.59(c). Subpart D, "Applications," of 10 CFR Part 55 requires that operator license applications must contain information about an individual's training and experience, unless the facility licensee certifies that the applicant has successfully completed a Commission-approved training program that is SAT-based and uses an acceptable simulation facility.

Both initial licensing and requalification training include training conducted on a control room simulator. Although the NRC does not mandate specific simulator training requirements (i.e., simulator training is determined by each facility licensee through the SAT process), typical initial licensing classes include 200 or more hours of simulator training, whereas requalification training includes 40 or more hours per year of simulator training. Simulator training includes normal integrated plant operations (e.g., startups, shutdowns, heat ups, cool downs, refueling, testing, technical specifications); abnormal, alarm, and transient response; and emergency response, including safety function challenges.

Associated with emergency response, operators and other plant staff are trained and examined on aspects of the facility's emergency plan, including requirements for maintaining sufficient staff during all modes of plant operation. For operating crews, routine emergency response training is conducted in the simulator using short (approximately 1-2 hour) scenarios. A facility's complete emergency response organization is exercised once every 2 years using scenarios lasting several hours during drills that the NRC observes.

The operator licensing process at power reactors includes a generic fundamentals examination covering the theoretical knowledge required to operate a nuclear power plant. License applicants must pass the generic fundamentals examination before they can take a site-specific examination. The site-specific examination consists of a written examination and an operating test that includes a plant walkthrough and a dynamic performance demonstration on a simulation facility.

The NRC staff has transferred most of the responsibility for developing site-specific licensing examinations to facility licensees. In 1999, the NRC amended 10 CFR Part 55 to allow nuclear power reactor licensees to prepare the written examinations and operating tests that the agency uses to evaluate the competence of applicants for operators' licenses at those facilities. Licensees that elect to prepare their own examinations are required to establish procedures to control examination security and integrity. They prepare and submit proposed examinations and operating tests to the NRC according to the guidance in NUREG-1021, Supplement 1, "Operator Licensing Examination Standards for Power Reactors," Revision 9, issued October 2007. The NRC reviews the facility-prepared examinations, prepares examinations for facility licensees upon request, administers all operating tests, makes the final licensing decisions, and issues the licenses.

As required by 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel," licensees must establish, implement, and maintain training programs using a SAT approach for eight categories of nonlicensed workers at nuclear power plants and for the shift supervisor, who is licensed in accordance with 10 CFR Part 55. These provisions complement the requirements for training based on a systems approach for the requalification of licensed operators and licensed senior operators. RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 3, issued May 2000, contains guidance to implement the regulations.

The NRC continues to endorse the training accreditation process that INPO manages. The staff recognizes that training programs developed in accordance with INPO guidelines and accredited by the National Nuclear Accrediting Board are SAT based; therefore, accredited programs are considered to be consistent with the regulations in 10 CFR Part 55 and 10 CFR 50.120. The NRC also recognizes that INPO-managed accreditation and associated training evaluation activities are an acceptable way of self-improvement in training. Such recognition encourages industry initiative and reduces NRC evaluation and inspection activities.

In accordance with its memorandum of agreement with INPO, the NRC monitors INPO accreditation activities as part of its continuing assessment of the effectiveness of the industry's training programs. Specifically, the NRC staff observes selected accreditation team visits and

NRC managers periodically observe National Nuclear Accrediting Board meetings. These observations are intended to monitor the implementation of programmatic aspects of the accreditation process, and they also give an opportunity to assess the selected performance areas of facility licensees.

If the National Nuclear Accrediting Board has concerns about the performance of an accredited training program, it will place the program on probation. This does not necessarily place a training program in noncompliance with either 10 CFR Part 55 or 10 CFR 50.120 because training programs are accredited to a standard of excellence rather than to a minimum level of regulatory compliance. However, the NRC does review the circumstances leading to the probation to ensure safe operations and continued compliance with the regulations.

The National Nuclear Accrediting Board may also withdraw accreditation in response to major deficiencies in a licensee's accredited training program. If accreditation is withdrawn, the NRC would ask that the licensee report the circumstances of the withdrawal for the staff to determine the significance of the issues related to the withdrawal. If the NRC determines that compliance with the regulations is not affected, it may not be necessary to take any further action. If the withdrawal is linked to a breakdown in the training process or a safety-significant issue, the NRC will conduct an immediate inspection focused on the process problem or safety issues. If appropriate, the agency would take further action, such as issuing confirmatory action letters or orders.

The NRC monitors industry performance in implementing the training requirements of 10 CFR Part 50 and 10 CFR Part 55 by (1) reviewing licensee event reports and inspection reports for training issues, (2) observing the accreditation process, and (3) reviewing the results of operator licensing activities. Guidance for periodically inspecting the licensed operator requalification training program at every facility is given in IP 71111.11, "Licensed Operator Requalification Program," dated August 27, 2012. When appropriate for cause, the NRC will also use IP 41500, "Training and Qualification Effectiveness," dated June 13, 1995, which references the guidance in NUREG-1220, "Training Review Criteria and Procedures," Revision 1, issued January 1993, to verify compliance with SAT requirements.

11.2.2 Experience

The NRC continually reviews operating experience information (e.g., event reports, inspection reports, reactor scrams, safety system actuations and failures, and forced plant outages) and monitors for trends concerning human performance, decisionmaking, and training, among other areas. Since the last CNS report was issued in 2010, there has been no notable increase in the trends associated with training deficiencies and operator errors. However, the NRC has noticed increased examples of nonconservative decisions that facility licensee personnel have made over the past few years, and the NRC has provided additional inspector guidance when reviewing certain decisions (i.e., equipment operability determinations) that facility licensees have made.

11.3 Fukushima Lessons Learned

There have been no changes in licensee financial resource considerations as a result of the Fukushima nuclear accident. Existing regulatory processes control and maintain the financial qualifications program for construction and operations, the financial qualifications program for decommissioning, the financial protection program for liability claims arising from accidents and the insurance program for onsite property damages arising from accidents.

On March 12, 2012, the NRC issued licensees an RFI that, in part, required licensees to reevaluate, analyze, and address, as necessary, postaccident emergency staffing. This RFI is discussed in detail in Section 16.9.

NRC financial and human resource considerations related to Fukushima are broken down in three categories, which are discussed in Section 8.3.

ARTICLE 12. HUMAN FACTORS

Each Contracting Party shall take the appropriate steps to ensure that the capabilities and limitations of human performance are taken into account throughout the life of a nuclear installation.

This section explains the NRC program on human performance. This program has seven major areas: (1) human factors engineering issues, (2) emergency operating procedures and plant procedures, (3) working hours and staffing, (4) fitness for duty, (5) Human Factors Information System, (6) support to event investigations and for-cause inspections, and (7) training. This section also discusses lessons learned from Fukushima.

12.1 Goals and Mission of the Program

The NRC has a comprehensive program for ensuring that human performance is properly addressed in a risk-informed regulatory framework for maintaining reactor safety. The NRC developed the program based on reviewing risk information and activities in the domestic and international nuclear industry.

12.2 Program Elements

The Reactor Oversight Process (discussed in Article 6) focuses on safety cornerstones that are assessed through a combination of performance indicators and risk-informed inspections that focus on risk-significant activities and systems related to the cornerstones. The three elements that cut across the cornerstones are human performance, a safety-conscious work environment, and problem identification and resolution. The Human Performance Program has contributed directly to the development of a supplemental IP related to the human performance cross-cutting element. The Human Performance Program is also engaged in the other two elements, as a safety-conscious work environment and many of the actions involved in corrective action programs result from human performance problems.

The Human Performance Program also supports the risk-informed and performance-based plan by generating, collecting, and evaluating data on human performance for use in human reliability analysis models. The staff evaluates information to gain insights supporting risk-informed regulation and to find human performance data for human reliability analysis. The NRC is working with industry to develop and implement the Scenario Authoring, Characterization, and Debriefing Application database to collect licensed operator simulator training and experimental data to support regulatory applications in human reliability analysis and human factors.

The Human Performance Program monitors technological developments and emerging issues to help prepare the NRC for the future. Because licensees are replacing aging analog controls and displays with digital components, the NRC must be prepared to review safety issues for human-system interfaces resulting from such new designs and technologies. The NRC has been processing many industry requests to transfer operating licenses, which may involve changes in organizational structure affecting human performance.

12.3 Significant Regulatory Activities

The NRC performs significant regulatory activities in the following seven areas to address human performance:

- human factors engineering issues
- emergency operating procedures and plant procedures
- working hours and staffing
- fitness for duty
- Human Factors Information System
- support to event investigations and for-cause inspections
- training

The following sections cover the first six activities; Article 11 describes training.

12.3.1 Human Factors Engineering Issues

This section discusses human factors activities related to engineering issues.

Governing Documents and Process. The NRC evaluates the human factors engineering design of the main control room and control centers outside of the main control room using NUREG-0800, Chapter 18, "Human Factors Engineering," Revision 2, issued March 2007, NUREG-0700, "Human System Interface Design Review Guideline," Revision 2, issued May 2002, and NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, issued November 2012. These documents provide guidance for the review of human-system interface issues in connection with the design certification of nuclear installations and the NRC's inspection program. The NRC also uses NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, issued September 2007, to review license amendment requests that credit the use of manual actions. Moreover, Information Notice (IN) 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," dated October 23, 1997, identifies references that the NRC uses to review the completion times of operator manual actions and how the actions will be reflected in the licensee's emergency procedures and operator training. In October 2007, the staff published NUREG-1852 "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," for use in evaluating exemptions from fire protection requirements that assume credit for timely manual actions.

In an effort to make some of the current human factors guidance simpler, clearer, and more relevant to the digital environment, the staff issued "Digital Instrumentation and Controls (DI&C) Interim Staff Guidance (ISG) DI&C-ISG-05, "Highly-Integrated Control Rooms—Human Factors Issues (HICR-HF) ," Revision 1, dated November 3, 2008, about computer-based procedures, minimum inventory of controls and displays to support plant shutdown, and crediting manual operator actions in diversity and defense-in-depth analyses. The crediting manual operator actions in diversity and defense-in-depth analyses interim guidance has been incorporated into permanent regulatory guidance through Appendix A, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses," of Chapter 18 to NUREG-0800. The NRC plans to endorse Institute of Electrical and Electronics Engineers (IEEE) standard 1786-2011, "IEEE Guide for Human Factors Applications of Computerized Operating Procedure Systems (COPS) at Nuclear Power Generating Stations and Other Nuclear Facilities," dated

September 22, 2011, by a regulatory guide (RG) that contains guidance pertaining to computer based procedures.

Experience. The NRC reviews licensees' requests that involve aspects of human factors engineering. Examples include crediting operator manual actions in amendments to plant technical specifications, transferring facility operating licenses, and increasing the reactor's authorized power level (i.e., power uprates). A recent license amendment request from Limerick Generating Station, Units 1 and 2, is an example of a review involving new or modified operator manual actions. The Limerick amendment proposed changes to manual actions as a result of a measurement uncertainty recapture and subsequent changes to the standby liquid control system requiring new operator manual actions.

The NRC has also evaluated some requests to transfer facility operating licenses, which affected management and organization, staffing, and technical qualifications. The NRC used NUREG-0800, Chapter 13, "Conduct of Operations," as the principal guidance for these reviews.

The NRC also reviews and approves requests for power uprates from currently licensed plants. For such requests, the NRC examines the effect of the power uprate on plant procedures, controls, displays, and alarms, and required operator actions using Section 2.11.1 or Review Standard (RS-001), "Review Standard for Extended Power Uprates," issued December 2003. The agency recently reviewed and approved power plant uprates for Grand Gulf Nuclear Station, and Turkey Point Nuclear Plants, Units 3 and 4.

12.3.2 Emergency Operating Procedures and Plant Procedures

Licensees must have programs to develop, implement, and maintain emergency operating and plant procedures. Article 16 discusses emergency preparedness; the discussion here is limited to the human factors aspect of emergency operating procedures.

Governing Documents and Process. On December 17, 1982, the NRC issued Generic Letter 1982-33, "Requirements for Emergency Response Capability," which transmitted NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," which requires each licensee to submit a set of documents for developing emergency operating procedures. In addition, Criterion V, "Instructions, Procedures, and Drawings," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, requires licensees to have operating procedures.

Experience. In 2010, a fire and subsequent complicated reactor trip at H.B. Robinson had complications in part because emergency operating procedures were inadequate. This resulted in the loss of reactor coolant pump seal cooling, which operators did not recognize. Inspection Report 261/2010-004 describes the violation and provides information on the event. A followup inspection (Inspection Report 261/2011-010) found that the plant's emergency operating procedures were structured in a nonstandard manner, and that, as part of the corrective actions, they would update the procedures to standard 2-column Westinghouse format.

On September 9, 2011, the NRC issued SECY-11-0124, "Recommended Actions To Be Taken without Delay From the Near Term Task Force Report," regarding lessons learned from Fukushima. The human factors staff is currently working with Recommendation 8 from that report, "strengthening and integration of emergency operating procedures, severe accident management guidelines, and extensive damage mitigation guidelines." Recommendation 8 is related to updating emergency operating procedures, SAMGs, and extensive damage mitigation guidelines to cope with beyond-design-basis accidents, such as the earthquake and tsunami that crippled Fukushima. During preparation of the draft regulatory basis, the staff described a lack of regulatory requirements in the areas of severe accident management guidelines and supporting procedures, training and qualification of personnel, exercises, and command and control structures associated with severe accidents. Furthermore, the staff determined that accident mitigating strategies are scattered throughout several sets of procedures developed through separate initiatives with minimal integration of the procedures to ensure cohesion and effectiveness.

The staff has drafted a regulatory basis that recommends developing a proposed rule that would require the licensees to integrate accident mitigating procedures, identify requirements for a severe accident command and control organization, and amend current rules for training and emergency exercises to include requirements related to severe accidents.

The staff held a public meeting on January 31, 2013, to give the public an opportunity to ask questions about the draft regulatory basis and an opportunity to exchange information on the proposed regulatory approach. The staff briefed the Advisory Committee on Reactor Safeguards on the draft regulatory basis on February 6, 2013. The public comment period on the draft regulatory basis closed on February 22, 2013. The final regulatory basis is scheduled to be completed by October 2013. The NTTF Recommendation 8 proposed rule is due July 2014, and the final rule is due in March 2016.

12.3.3 Shift Staffing

Governing Documents and Process. In 10 CFR 50.54(m), the NRC specifies the minimum number of licensed operators and senior operators required for nuclear power reactor facilities. Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” and Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50 contain the NRC staffing requirements for fire brigades and emergency response personnel.

In September 2002, the NRC began work on a process to evaluate exemption requests from the requirements in 10 CFR 50.54(m) resulting from the changing demands and new technologies presented by advanced reactor control room designs and significant light-water reactor control room upgrades. In July 2005, the NRC issued NUREG-1791, “Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m).” The purpose of reviewing the exemption requests is to ensure public health and safety by verifying that the applicant’s staffing plan and supporting analyses sufficiently justify the requested exemption. NUREG/CR-6838, “Technical Basis for Assessing Exemptions from Nuclear Power Plant Licensed Operator Staffing Requirements in 10 CFR 50.54(m),” issued February 2004, explains the justification for the recommended process.

SECY-10-0034, “Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs,” issued March 2010, discusses appropriate requirements for operator staffing for small or multimodule (advanced reactor) facilities. Current regulations do not address the possibility of more than two reactors being controlled from one control room. In addition, small modular reactor designers have indicated that they are considering whether their designs can operate with a staffing complement that is less than what the Commission regulations currently require. Other small modular reactors policy issues include the possible need for requirements on control room staffing during refueling operations, reactor staff that interact with an interconnected manufacturing plant, supervisory staff, shift work, and training. The NRC staff has stated in previous reports that it believes that operator crew staffing may be design dependent and intended to review the justification for a smaller crew size for the advanced reactors by evaluating the function and task analyses for normal operation and accident management. Should it be necessary, the staff will propose changes to existing RGs or staff positions or propose new guidance concerning the operator staffing for a small modular reactor to support development of the Next Generation Nuclear Plant and other small modular reactor designs.

Experience. No significant examples of shift staffing issues were identified for 2011-2012.

12.3.4 Fitness for Duty

This section discusses the NRC's requirements pertaining to the fitness for duty of nuclear power plant workers, including requirements regarding drug and alcohol testing, behavioral observation and management of worker fatigue.

Governing Documents and Process. As required by 10 CFR Part 26, "Fitness for Duty Programs," each licensee authorized to operate or construct a nuclear power reactor must implement a fitness for duty program for all personnel who have unescorted access to the protected area of its plant or who perform the duties specified in 10 CFR 26.4 (e.g., fitness for duty program personnel and emergency response personnel). Licensees and permit holders authorized to construct a nuclear power plant must implement a fitness for duty program for personnel performing certain construction, management, security, and quality control activities. For performance objectives, 10 CFR Part 26 requires that licensees establish programs that (1) give reasonable assurance that nuclear power plant personnel perform their tasks in a reliable and trustworthy manner as demonstrated by the avoidance of substance abuse, (2) provide reasonable assurance that persons are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, (3) provide reasonable measures for the early detection of persons who are not fit to perform activities, (4) provide reasonable assurance that workplaces subject to 10 CFR Part 26 are free from the presence and effects of illegal drugs and alcohol and, (5) provide reasonable assurance that nuclear power plant management are managing the effects of fatigue on an individual's ability to safely and competently perform his or her duties.

The NRC issues annual reports on statistical data and lessons learned from licensee's fitness for duty program performance reports. The most recent of these is "Summary of Fitness-for-Duty Program Performance Reports for Calendar Year 2011," located at http://pbadupws.nrc.gov/docs/ML1215/ML12151A270.pdf. A project to enable electronic reporting of performance data has been implemented and has resulted in a substantial improvement in data reporting and evaluation as reflected in the 2011 performance report.

For worker fatigue, on March 31, 2008, the NRC published a rule that included new regulations in 10 CFR Part 26, Subpart I, "Managing Fatigue." Subpart I strengthens the effectiveness of fitness for duty programs for protecting public health and safety by establishing enforceable requirements for the management of worker fatigue. In addition to the rulemaking and its associated analyses, the NRC issued RG 5.73, "Fatigue Management for Nuclear Power Plant Personnel," in March 2009, to provide guidance on how to implement the rule.

Experience. Following implementation of the rule, the NRC received several petitions for rulemaking to alleviate alleged impacts adverse to safety that were introduced when the rule was implemented. The petitioners alleged that implementation of the rule had impeded some beneficial safety practices. The NRC worked with the industry and other external stakeholders to develop an alternative method for managing cumulative fatigue. The alternative method limits work hours to a weekly average of 54 hours worked, with work hours being averaged over a rolling period of up to 6 weeks. As a result, the alternative method limits work hours to levels comparable to the original requirements while adding the simplicity and flexibility desired by the industry. The rule codifying the alternative method was published on July 21, 2011, and the rule was effective on August 22, 2011. To date, several licensees have adopted the alternative method and feedback indicates that it has allowed the beneficial safety practices to be reinstituted at those facilities that adopted that alternative. Additional information can be found in Section 1.3.3 of this report.

12.3.5 Human Factors Information System

Governing Documents and Process. The Human Factors Information System is designed to store, retrieve, sort, and analyze human performance information extracted from NRC inspection and licensee event reports. Initiated in 1990, this automated information management system can generate a variety of specialized reports that are not readily available from other NRC sources. In 2006, the NRC improved this system to better align the coding scheme with the Reactor Oversight Process and to enhance the system's search capabilities. The Human Factors Information System now captures information related to training, procedures and reference documents, fitness for duty, oversight, problem identification and resolution, communications, human-system interface and environment, and work planning and practices.

Experience. The NRC responds to stakeholder and public inquiries and data requests on this system on a regular basis. For example, inspectors use the data this system generates in preparing inspection activities related to human performance. In addition, the NRC's Office of Nuclear Regulatory Research uses the data to support activities in human performance and human reliability analysis. Other NRC program offices use the data to gain insights about human performance, to monitor the frequency of human performance issues, and to inform several types of reports, such as internal operating experience reports. The NRC also uses a Web site to disseminate information on human performance issues at individual nuclear power plant sites.

12.3.6 Support to Event Investigations and For-Cause Inspections and Training

Governing Documents and Process. NRC staff members with human factors expertise often participate in special inspections, incident investigation team inspections, augmented team inspections, event investigations, and supplemental inspections. Human factors experts have assessed management effectiveness, procedures, training issues, staffing issues, human-machine interfaces, personnel performance issues, safety-conscious work environment, and safety culture.

For training issues, inspectors use IP 41500, "Training and Qualification Effectiveness," dated June 13, 1995. For procedure issues, inspectors use IP 42001, "Emergency Operating Procedures," dated June 28, 1991, and IP 42700, "Plant Procedures," dated November 15, 1995. For baseline inspections under the Reactor Oversight Process, inspectors use IP 71152, "Problem Identification and Resolution," dated January 31, 2013, which is intended to establish confidence that each licensee is detecting and correcting problems in a way that limits the risk to the public and includes a review of the licensee's safety-conscious work environment. A key premise of the Reactor Oversight Process is that weaknesses in problem identification and resolution programs will manifest themselves as performance issues that can be identified during the baseline inspection program or by crossing predetermined indicator thresholds.

For supplemental inspections, IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," as revised in February 2011, includes requirements for the NRC staff to review the licensee's third-party safety culture assessment and independently assess the licensee's safety culture. Staff members with technical expertise in human factors and safety culture perform the safety culture assessment activities. The NRC first implemented the revised IP 95003 at the Palo Verde Nuclear Generating Station in October 2007. Based on the lessons learned from the 2007 NRC inspection and on input from the industry and the public, the staff updated Inspection Manual Chapter 0305, "Operating Reactor Assessment Program," in 2009 and again in 2012.

Experience. In 2007, NRC staff with human factors expertise participated in an IP 95003 inspection at Palo Verde to assess human performance at the site. The inspectors determined that some findings related to procedure adherence had strong human performance contributions. The NRC discussed its safety concerns, and how and when these issues were identified with Palo Verde. Palo Verde made a commitment to take action to improve their performance.

The NRC increased its plant oversight and conducted numerous inspections. The results of these inspections demonstrated that performance at Palo Verde had improved substantially. In March 2009, the NRC determined that the commitments that Palo Verde previously made had been completed and decided to reduce its oversight at this site.

The NRC continued to monitor Palo Verde to verify that the facility is operating safely and that the licensee's performance improvements are being sustained by focusing on the effectiveness of the site's programs and processes. The NRC performed additional inspection activities in selected areas over a 2-year period to monitor Palo Verde improvement initiatives and to look for any indications of potential decline in safety performance at the site. The first of these inspections was performed in January 2010 to assess the effectiveness of the licensee's corrective actions in addressing the human performance issues identified during the IP 95003 inspection. The results of this inspection can be found in "Palo Verde Nuclear Generating Station—NRC Integrated Inspection Report Nos. 05000528/2010002, 05000529/2010002, and 05000530/2010002," dated May 5, 2010. Subsequent inspections continued to evaluate the performance initiatives and safety performance during the site's recovery period. These reports can all be found on the NRC's public Web site.

Since the Palo Verde inspection, the NRC human factors staff have participated in IP 95002 inspections at Browns Ferry Nuclear Station and Perry Nuclear Generating Station. The IP 95002 inspections are designed to evaluate the licensees' root cause analyses of the findings and violations to ensure that they have identified the correct safety culture components and have developed corrective actions to prevent recurrence. In addition, the NRC staff recently completed an IP 95003 at Fort Calhoun Nuclear Station and is preparing to perform an IP 95003 inspection at Browns Ferry Nuclear Station in spring 2013. The NRC will continue to incorporate lessons learned from these inspections.

12.4 Fukushima Lessons Learned

There are human factors considerations to many of the Fukushima lessons learned, including the orders issued in March 2012, which require nuclear power plant operators to implement safety enhancements related to: (1) mitigation strategies to respond to external events resulting in the loss of power at plants, (2) ensuring reliable hardened containment vents for BWR Mark I and II designs, and (3) enhancing SFP instrumentation. The plant operators are required to begin implementation of the safety enhancements promptly, including consideration of human factors, and complete implementation within two refueling outages or by December 31, 2016, whichever comes first.

Human factors are also considered as part of the RFI issued by the NRC in March 2012, specifically the request for each licensee to assess its emergency communications systems and staffing levels for responding to an accident

NRC financial and human resource considerations related to Fukushima are broken down into three categories, which are discussed in Section 8.3.

ARTICLE 13. QUALITY ASSURANCE

Each Contracting Party shall take the appropriate steps to ensure that quality assurance programmes are established and implemented with a view to providing confidence that specified requirements for all activities important to nuclear safety are satisfied throughout the life of a nuclear installation.

This section describes quality assurance requirements and guidance for design and construction, operational activities, and staff licensing reviews. It also describes quality assurance programs, regulatory guidance, and lessons learned from Fukushima.

13.1 Background

Nuclear power facilities must be designed, constructed, and operated in a manner that ensures: (1) the prevention of accidents that could cause undue risk to public health and safety and (2) the mitigation of adverse consequences of such accidents if they should occur. A primary way to achieve these objectives is to establish and effectively implement a nuclear quality assurance program. Although a licensee may delegate aspects of the establishment or execution of the quality assurance program to others, the licensee remains ultimately responsible for the program's overall effectiveness. Licensees carry out a variety of self-assessments to validate the effectiveness of their quality assurance program implementation. The NRC reviews descriptions of quality assurance programs and performs onsite inspections to verify aspects of the program implementation.

13.2 Regulatory Policy and Requirements

The NRC sets forth requirements for a license to design, construct, and operate commercial nuclear power plants in both 10 CFR Part 50 and 10 CFR Part 52. Specifically, 10 CFR Part 50 contains the requirements for a construction permit and a separate operating license and 10 CFR Part 52 includes the requirements for a single combined license, which allows for both construction and operation of a nuclear power plant.

For either type of license, an applicant must describe its quality assurance program for all activities affecting the safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to public health and safety. High-level criteria for determining which plant SSCs are safety-related appear in 10 CFR 50.2, "Definitions." Based on these criteria, licensees' engineering organizations develop plant-specific listings of safety-related SSCs.

Under the 10 CFR Part 50 licensing process, each applicant for a construction permit must describe its quality assurance program in its preliminary safety analysis report in accordance with 10 CFR 50.34(a)(7). This program should apply to the design, fabrication, construction, and testing of SSCs. In accordance with 10 CFR 50.34(b)(6)(ii), each applicant for an operating license under 10 CFR Part 50 must describe the managerial and administrative controls that will be implemented during the operation of the nuclear power plant. The applicant must also describe how it will satisfy the applicable requirements of Appendix B to 10 CFR Part 50.

Each applicant for a combined license under 10 CFR Part 52 must describe its quality assurance program in a safety analysis report and give a description of the managerial and administrative controls that will be implemented during the operation of the nuclear power plant. Like a 10 CFR Part 50 applicant, an applicant under 10 CFR Part 52 must also describe how it will satisfy the applicable requirements of Appendix B to 10 CFR Part 50.

13.2.1 Appendix A to 10 CFR Part 50

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 details the general requirements for establishing quality assurance controls. General Design Criterion 1, "Quality Standards and Records," contains requirements that apply to the quality assurance of items important to safety. The scope of items that are "important to safety" includes a subset of plant equipment classified as safety-related. Appendix B to 10 CFR Part 50 (discussed in Section 13.2.2 of this report) contains quality assurance program requirements for safety-related SSCs. Other regulatory guidance discusses quality assurance program controls that are appropriate for some types of nonsafety-related equipment.

13.2.2 Appendix B to 10 CFR Part 50

Appendix B to 10 CFR Part 50 outlines the quality assurance requirements that apply to activities affecting the safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents. Appendix B defines quality assurance as all planned and systematic actions that are necessary for adequate confidence that SSCs will perform satisfactorily in service. Toward that end, it specifies 18 criteria that the commitments in a licensee's quality assurance program must satisfy. These criteria cover such topics as organizational independence, design control, procurement, document control, test control, corrective action, and audits. Appendix B also stipulates that licensees establish measures to ensure that the documents for procurement of safety-related materials, equipment, and services, whether purchased by the licensee or its contractors or subcontractors, include or reference the applicable regulatory requirements, design bases, and other requirements necessary to ensure adequate quality. Consistent with the importance and complexity of the products or services to be provided, licensees (or their designees) are responsible for periodically verifying that suppliers' quality assurance programs comply, as appropriate, with the applicable criteria in Appendix B and that they are effectively implemented. Additionally, as outlined in 10 CFR 21.41, "Inspections," the NRC staff performs inspections at vendors that supply basic components to the nuclear industry.

Because the requirements of Appendix B are written at a conceptual level, the NRC and the industry needed to develop consensus standards that include acceptable ways to conform to these requirements. The NRC then issued companion RGs, which endorsed (with conditions, if warranted) quality assurance codes and standards.

13.2.3 Approaches for Adopting More Widely Accepted International Quality Standards

The NRC has reviewed options for adopting more widely accepted international quality standards, such as International Organization for Standardization Standard 9001, 2000 edition, by considering how international standards compare with the existing framework in Appendix B to 10 CFR Part 50. On the basis of this review, the NRC concluded that supplemental quality requirements would be needed when implementing Standard 9001 within the existing regulatory framework. The NRC participates in both national and international efforts associated with quality assurance standard development and it continues to assess how various national and international quality standards comport with NRC regulations in an ongoing effort to seek convergence of standards.

13.3 Quality Assurance Regulatory Guidance

The NRC has developed or endorsed quality assurance guidance for use by the NRC staff, applicants for construction permits or operating licenses, and licensees. This guidance is applicable to the design, construction, and operational phases of a nuclear power plant.

13.3.1 Guidance for Staff Reviews for Licensing

NUREG-0800, Section 17.5, "Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants," issued March 2007, provides guidance to the NRC staff for the review of applications for construction permits, operating licenses, and combined licenses. The specific review guidance in NUREG-0800 correlates with the 18 criteria of Appendix B to 10 CFR Part 50, and integrates a review of licensee commitments to adopt the NRC's quality assurance-related RGs and apply the industry's quality assurance codes and standards.

13.3.2 Guidance for Design and Construction Activities

Licensees may apply consensus standards developed by the American National Standards Institute (ANSI) in its N45.2 series or by ASME in its NQA-1 series to comply with the requirements of Appendix B to 10 CFR Part 50. The NRC has endorsed ANSI and ASME standards through its RGs. Through its consensus codes and standards activities, the NRC continues to participate with ASME NQA-1 committees to revise the latest edition of the NQA-1 standard. As part of this effort, the NRC staff issued RG 1.28, "Quality Assurance Program Requirements (Design and Construction)," Revision 4, on June 2010, to endorse NQA-1-2008 and the NQA-1a-2009 addenda.

13.3.3 Guidance for Operational Activities

The NRC has conditionally endorsed the consensus standard ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," issued February 1976, through RG 1.33, "Quality Assurance Program Requirements (Operations)," Revision 2, issued February 1978, as complying with the requirements of Appendix B to 10 CFR Part 50. The NRC staff is planning to issue a revision to RG 1.33 to endorse ANSI/ANS 3.2-2012, "Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants." ANSI/ANS 3.2-2012 is an update to ANS 3.2/ANSI N18.7-1976 and incorporates operational experience since the original standard was developed, and is better focused on quality assurance of plant operations because information on quality assurance of design and construction was moved to another standard.

13.4 Quality Assurance Programs

The NRC inspects quality assurance programs under the Reactor Oversight Process for operating reactors and under the Construction Inspection Program (see Article 18 of this report) for new reactors. The NRC also conducts augmented inspection activities as needed.

The baseline inspection program of the Reactor Oversight Process includes one primary procedure related to quality assurance issues, IP 71152. Inspectors use this procedure to assess the effectiveness of licensees' programs to find and resolve problems through a performance-based review of specific issues. In particular, inspectors look for cases in which a licensee may have missed generic implications of specific problems and for the risk significance of combinations of problems that individually may not have significance. They do not inspect other aspects of quality assurance program implementation in the baseline inspection program but may do so through supplemental inspections.

Some equipment in the nuclear facility may be classified as nonsafety-related and yet still be important to safety for some unique reason. In specific cases, the NRC has specified that quality assurance controls are warranted for equipment determined to be more important than commercial-grade equipment. However, the quality assurance controls do not have to meet Appendix B requirements, which apply only to activities affecting safety-related functions. Typically, applying quality assurance controls to this important-to-safety, yet nonsafety-related, equipment is called "augmented quality control."

The Construction Inspection Program provides oversight for nuclear plants licensed under 10 CFR Part 50 and 10 CFR Part 52, including quality assurance program inspection. The quality assurance inspection program focuses on an applicant or licensee establishing and implementing a quality assurance program in accordance with the requirements of Appendix B to 10 CFR Part 50. The inspectors use IP 35007, "Quality Assurance Program Implementation during Construction and Pre-Construction Activities," dated January 15, 2013, to verify that the holder of a combined license has developed quality assurance procedures, instructions, and other documents that are consistent with the licensee's NRC-approved quality assurance program description, and to verify that the licensee has effectively implemented its quality assurance program implementing documents during construction activities.

As provided in the Construction Inspection Program, the nuclear plant will transition from the Construction Inspection Program to the Reactor Oversight Process for commercial operation when, in accordance with 10 CFR 52.103(g), the Commission determines that all of the inspections, tests, and analyses in the combined license have been performed, and the associated acceptance criteria have been met.

13.5 Quality Assurance Audits Performed by Licensees

Appendix B to 10 CFR Part 50 requires licensees to verify the effectiveness of their quality assurance program by performing internal audits of their programs. These audits are performed in accordance with the licensee's procedures by appropriately trained and qualified personnel who do not have direct responsibility for performing the activities being audited. The results of these audits are documented and given to management for review and corrective action.

13.5.1 Audits of Vendors and Suppliers

Appendix B to 10 CFR Part 50 requires licensees that procure material, equipment, or services from contractors or subcontractors to perform audits to ensure that suppliers implement an effective quality assurance program, consistent with the requirements of Appendix B and the licensee's technical requirements.

Licensees perform these activities by using their own technical and quality assurance staff. Industry initiatives to promote effective and efficient standardization of these audit activities have resulted in licensees sharing their technical resources through joint audits of suppliers.

13.6 Fukushima Lessons Learned

As a result of the Fukushima accident, there have been no changes to: (1) the regulatory and policy requirements of Appendix A and Appendix B to 10 CFR Part 50, (2) the quality assurance regulatory guidance, or (3) licensee quality assurance programs.

However, continued compliance with existing quality assurance programs and requirements is an important part of implementation of the lessons learned from Fukushima, including the orders issued in March 2012, which require nuclear power plant operators to implement safety enhancements related to: (1) mitigation strategies to respond to external events resulting in the loss of power at plants, (2) ensuring reliable hardened containment vents for BWR Mark I and II designs, and (3) enhancing SFP instrumentation. The plant operators are required to begin implementation of the safety enhancements promptly and complete implementation within two refueling outages or by December 31, 2016, whichever comes first.

Quality assurance is also an important part of the responses to the RFI issued by the NRC in March 2012, which requested each licensee to reevaluate the seismic and flooding hazards at its site using present-day methods and information, conduct inspections (or “walkdowns”) of existing seismic and flood protection features, and assess its emergency communications systems and staffing levels.

ARTICLE 14. ASSESSMENT AND VERIFICATION OF SAFETY

Each Contracting Party shall take the appropriate steps to ensure that:

(i) comprehensive and systematic safety assessments are carried out before the construction and commissioning of a nuclear installation and throughout its life. Such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information, and reviewed under the authority of the regulatory body

(ii) verification by analysis, surveillance, testing, and inspection is carried out to ensure that the physical state and the operation of nuclear installations continue to be in assurance with its design, applicable national safety requirements, and operational limits and conditions

This section explains the governing documents and process for ensuring that systematic safety assessments are carried out during the life of the nuclear installation, including for power uprates and the period of extended operation. It focuses on assessments performed to maintain the licensing basis of a nuclear installation. Finally, this section explains verification of the physical state and operation of the nuclear installation by analysis, surveillance, testing, and inspection, and lessons learned from Fukushima.

Other articles in this report (e.g., Articles 6, 10, 13, 18, and 19) also discuss activities to achieve safety at nuclear installations.

14.1 <u>Ensuring Safety Assessments throughout Plant Life</u>

Before a nuclear facility is constructed, commissioned, and licensed, an applicant must perform comprehensive and systematic safety assessments for NRC review and approval. Article 18 of this report discusses these assessments and reviews.

This section focuses on the assessments required throughout the life of a nuclear installation (i.e., assessments required to maintain the licensing basis). To show conformance with the licensing basis, a licensee must maintain records of the original design bases and any changes. This section explains how such changes are documented, updated, and reviewed. Renewal of a license depends on a licensee's continuing to meet its current licensing basis; this section explains how the license renewal process accounts for this requirement.

14.1.1 Assessment of Safety

The Reactor Oversight Process is the NRC's program to inspect, measure, and assess the safety and security performance of commercial nuclear power plants. The objective of the Reactor Oversight Process is to monitor reactor performance in three key areas (i.e., reactor safety, radiation safety, and safeguards), which are subsequently monitored through seven cornerstones. The Reactor Oversight Process assesses plant performance using both inspection findings and performance indicators across the seven cornerstones. The NRC determines its regulatory response to plant performance in accordance with an Action Matrix that provides for a range of actions commensurate with the significance of the inspection findings and performance indicators. The Action Matrix is intended to provide consistent, predictable and understandable agency responses to licensee performance such that the NRC's regulatory oversight increases as licensee performance declines.

Section 6.3.2 of this report discusses the Reactor Oversight Process and results of regulatory assessment in greater detail.

The Construction Reactor Oversight Process monitors and assesses the construction of commercial nuclear power plants in a similar manner to that employed by the Reactor Oversight Process. The NRC monitors plant construction in three key areas (i.e., construction reactor safety, operational readiness, and safeguards programs) and assesses construction using inspection findings across six cornerstones. The NRC determines its regulatory response to licensee construction performance in accordance with the Construction Action Matrix.

14.1.2 Maintaining the Licensing Basis

The NRC carries out regulatory programs to give reasonable assurance that plants continue to conform to the licensing basis. Article 6 of this report discusses these programs.

This section explains the governing documents and process used to maintain the licensing basis, as required by 10 CFR 50.90, "Application for Amendment of License, or Construction Permit, or Early Site Permit," 10 CFR 50.59, "Changes, Tests and Experiments," and 10 CFR 50.71, "Maintenance of Records, Making of Reports."

14.1.2.1 Governing Documents and Process

A licensee is to operate its facility in accordance with the license and as described in its final safety analysis report. To change its license or reactor facility, a licensee must follow the review and approval processes established in the regulations. For license amendments, including changes to technical specifications, the licensee must ask for NRC approval in accordance with 10 CFR 50.90. However, 10 CFR 50.59 contains requirements for the process by which, under certain conditions, licensees may make changes to their facilities and procedures as described in the safety analysis report without prior NRC approval.

10 CFR 50.59. In 10 CFR 50.59, the NRC establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval. The NRC must review and approve proposed changes, tests, and experiments that satisfy the definitions and one or more of the criteria in the rule before implementation. Thus, the rule provides a threshold for regulatory review, not the final determination of safety, for proposed activities. After determining that a proposed activity is safe and effective through appropriate engineering and technical evaluations, the 10 CFR 50.59 process is applied to determine if a license amendment will be required before implementation. The process involves three basic steps: (1) applicability and screening to determine if a 10 CFR 50.59 evaluation is required, (2) an evaluation that applies the eight evaluation criteria of 10 CFR 50.59(c)(2) to determine if a license amendment must be obtained from the NRC, and (3) documentation and reporting to the NRC of activities implemented under 10 CFR 50.59.

A licensee shall obtain a license amendment in accordance with 10 CFR 50.90 before implementing a proposed change, test, or experiment if the change, test, or experiment would do any of the following:

- result in more than a minimal increase in the frequency of occurrence of a previously evaluated accident
- result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety
- result in more than a minimal increase in the consequences of a previously evaluated accident
- result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety
- create a possibility for an accident of a different type than any previously evaluated
- create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated
- result in exceeding or altering a design-basis limit for a fission product barrier
- result in a departure from a method of evaluation used in establishing the design bases or in the safety analyses

10 CFR 50.90. According to 10 CFR 50.90, whenever a holder of a license, including a construction permit and operating license under 10 CFR Part 50, or an early site permit, combined license, or manufacturing license under 10 CFR Part 52, wants to amend the license or permit, it must file an application for an amendment with the Commission, as specified in 10 CFR 50.4, "Written Communications," or 10 CFR 52.3, "Written Communications," fully describing the changes desired, and following, as far as applicable, the form prescribed for original applications. The NRC performs and documents a safety evaluation in these instances before it authorizes the change.

10 CFR 50.71. In Section e of 10 CFR 50.71, "Maintenance of Records, Making of Reports," the NRC describes another process for making changes. This regulation requires licensees to update their final safety analysis reports periodically to incorporate the information and analyses that they submitted to the Commission or prepared in accordance with Commission requirements. Revisions to the updated final safety analysis reports are to include the effects of changes that occur in the vicinity of the plant, changes made in the facility or procedures described in the report, safety evaluations for approved license amendments and for changes made under 10 CFR 50.59, and safety analyses conducted at the request of the Commission to address new safety issues.

14.1.3 Power Uprates

This section explains the NRC power uprate licensing process, including the governing documents, regulatory process, recent experience, and relevant examples.

14.1.3.1 Governing Documents and Process

Background. The NRC regulates the maximum power level at which a commercial nuclear power plant may operate. This power level is used, with other data, in many of the licensing analyses that demonstrate plant safety. This power level is included in the license and technical specifications for the plant. NRC approval is required to make changes to the license and technical specifications for a plant. Thus, a licensee must receive NRC approval, through the license amendment process, before it can operate at a higher power level.

Categories of Power Uprates. The NRC has specified three categories of power uprates:

- Measurement Uncertainty Recapture Power Uprates - measurement uncertainty recapture power uprates are power increases of less than 2 percent and are achieved by implementing enhanced techniques for calculating reactor power. This involves the use of state-of-the-art devices to more precisely measure feedwater flow that is used to calculate reactor power. More precise measurements reduce the degree of uncertainty in the power level, which analysts use to predict the ability of the reactor to be safely shut down under postulated accident conditions.

- Stretch Power Uprates - stretch power uprates typically are on the order of up to 7 percent and are within the design capacity of the plant. The actual value for percentage increase in power a plant can achieve and stay within the stretch power uprate category is plant-specific and depends on the operating margins included in the design of a particular plant. Stretch power uprates usually involve changes to instrumentation setpoints but do not involve major plant modifications.

- Extended Power Uprates - extended power uprates are greater than stretch power uprates and have been approved for increases as high as 20 percent. Extended power uprates usually require significant modifications to major balance-of-plant equipment such as the high pressure turbines, condensate pumps and motors, main generators, and/or transformers.

Review Process, Regulatory Requirements, and Guidance Documents. Since uprates affect a reactor's licensed power level, utilities apply for NRC permission to amend their operating license to implement a power uprate. The process for requesting and approving a change to a plant's power level is governed by 10 CFR 50.90 through 10 CFR 50.92, "Issuance of Amendment." The applications and reviews are often complex and involve many areas of expertise in the NRC's Office of Nuclear Reactor Regulation and Office of the General Counsel. Some reviews also may involve the Office of Nuclear Regulatory Research, Office of New Reactors, and the Advisory Committee on Reactor Safeguards. In evaluating a power uprate request, NRC reviews data and accident analyses that a licensee submits to confirm that the plant can operate safely at the higher power level.

The NRC uses RS-001, "Review Standard for Extended Power Uprates," issued December 2003, for evaluating extended power uprates and stretch power uprates. The Advisory Committee on Reactor Safeguards has endorsed this standard, which provides a comprehensive process and technical guidance for reviews by the NRC staff, and useful information to licensees considering applying for an extended power uprate. RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," issued January 2002, discusses the scope and detail of the information that should be provided to the NRC for reviewing measurement uncertainty recapture uprate applications. Additionally, the staff uses NUREG-0800, where appropriate, when conducting power uprate regulatory reviews.

After a licensee submits an uprate application, the NRC issues a *Federal Register* notice to alert the public that the agency is considering the application. The public has 30 days to comment on the licensee's request and 60 days to request a hearing where the application could be contested. The NRC thoroughly reviews the application and any public comments, while the Atomic Safety and Licensing Board considers any requests for hearings. The NRC documents its review in a safety evaluation, and, if acceptable, the NRC will issue a license amendment approving the power uprate. The NRC will issue another *Federal Register* notice to inform the public if the amendment is issued. Following the approval, the NRC performs inspections of the power uprate implementation using Inspection Procedure 71004, "Power Uprates," dated April 30, 2010, to review plant modifications and operator readiness.

If the Atomic Safety and Licensing Board determines that a hearing is required, a separate legal process takes place, and NRC staff provides technical information, if needed. The safety evaluation and any hearing rulings form the basis for the NRC's final decision on the uprate request. However, the staff can authorize an uprate before the hearing is completed. The NRC issues a press release for any approved uprate.

The NRC's current schedule is to complete power uprate reviews within 18 months of application review acceptance for extended power uprates, 12 months of application review acceptance for stretch power uprates, and 9 months of application review acceptance for measurement uncertainty recapture uprates. The application acceptance process is intended to provide the NRC staff an opportunity to ensure that application quality is sufficient for regulatory review such that these schedules can be met.

14.1.3.2 Experience

The NRC issued the first power uprate amendment for the Calvert Cliffs nuclear power plant in 1977. As of August 2013, the NRC had approved 148 uprates, resulting in a gain of approximately 20,586 MWt (megawatts thermal) or 6,862 MWe (megawatts electric), at existing plants. The NRC is currently reviewing 14 power uprate applications that would authorize an additional 3,000 MWt or approximately 1,000 MWe. In addition, based on responses to an NRC survey issued on December 2012, licensees plan to submit 2 power uprate applications in the next 5 years, including 2 measurement uncertainty recapture power uprates. If these expected applications are approved, the resulting uprates would authorize an additional 114 MWt (38 MWe).

14.1.4 License Renewal

This section explains license renewal, including the governing documents, regulatory process, recent experience, and relevant examples.

14.1.4.1 Governing Documents and Process

Background. The Atomic Energy Act and NRC regulations limit commercial power reactor licenses to 40 years but permit such licenses to be renewed. The original 40-year term was selected on the basis of economic and antitrust considerations, not technical limitations.

The NRC has established a license renewal process that can be completed in a reasonable time period and has clear requirements to ensure safe plant operation for up to 20 additional years of plant life. The NRC's current schedule is to complete renewal reviews within 30 months of receipt of the application if a hearing is conducted, and within 22 months if a hearing is not conducted. As of February 2013, nine[4] license renewal applications are under NRC review. The Commission has decided that no final licenses will be issued until a new Waste Confidence Decision and Rule are in effect; however, the Commission directed the staff to advance licensing

4 Duke Energy withdrew its Crystal River Nuclear Generating Plant, Unit 3, license renewal application on February 6, 2013, reducing the number of license renewal applications under NRC review from 10 to 9. See Section 6.2 for additional information on Crystal River Unit 3.

reviews and proceedings to the extent practical while a new environmental impact statement and rule are under development. The decision to seek license renewal rests entirely with nuclear power plant owners and typically is based on the plant's economic situation and whether it can meet NRC requirements.

Research has concluded that aging phenomena are readily manageable and do not pose technical issues that would prevent life extension for nuclear power plants. Studies have also found that facilities deal adequately with many aging effects during the initial license period, and that credit should be given for these existing programs, particularly those under the NRC's Maintenance Rule, 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which helps manage plant aging.

The license renewal process proceeds along two tracks: one for the review of safety issues and another for environmental issues. An applicant must give the NRC an evaluation that addresses the technical aspects of plant aging and describes the ways it will manage those effects. It must also prepare an evaluation of the potential impact on the environment if the plant operates for up to 20 more years. The NRC reviews the application and verifies the safety and environmental issues through onsite audits and inspections. The NRC documents its findings in a safety evaluation report and an environmental impact statement.

Public participation is an important part of the license renewal process. Members of the public have opportunities to comment on the environmental review and question how aging will be managed during the period of extended operation. All information related to the review and approval of a renewal application is publicly available. Significant safety and environmental concerns also may be litigated in an adjudicatory hearing if any party that would be adversely affected asks for a hearing.

10 CFR Part 54. Known as the License Renewal Rule, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," establishes the technical and procedural requirements for renewing operating licenses. License renewal requirements for power reactors are based on two key principles:

(1) When continued into the extended period of operation, the regulatory process, which assesses and verifies safety, is adequate to ensure that the licensing basis of all currently operating plants provides an acceptable level of safety. The possible exception is detrimental effects of aging on certain SSCs, and possibly a few other issues applying to safety only during the period of extended operation.

(2) Each plant must maintain its licensing basis throughout the renewal term.

Guidance that applies to license renewal includes RG 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," Revision 1, issued September 2005, to help applicants apply to renew a license; and NUREG-1800, "Standard Review Plan for Review of License Renewal of Applications for Nuclear Power Plants," Revision 2, issued December 2010, which guides the staff in reviewing applications. The standard review plan for license renewal incorporates by reference NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, issued December 2010, which generically documents the basis for determining when existing programs are adequate for license renewal and when they should be augmented. As lessons are learned from the review of renewal applications or generic technical issues are resolved, the NRC issues improved guidance for interim use by applicants until the guidance is incorporated into the next formal update of the documents. The staff is currently preparing a project to revise both the standard review plan for license renewal and the GALL Report.

10 CFR Part 51. The NRC's environmental protection regulation, 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," also applies to license renewal of nuclear power plants. The agency amended this regulation to facilitate its environmental review process for license renewal while ensuring that its commitment to public engagement and transparency is maintained. The NRC performs plant-specific reviews of the environmental impacts of license renewal to determine whether the effects are so severe to preclude license renewal as an option for energy-planning decisionmakers.

The license renewal environmental review requirements under 10 CFR Part 51 are founded on the conclusion that certain environmental issues can be resolved generically and need not be evaluated in each plant-specific application. These issues are listed in Table B-1 of Appendix B, "Environmental Effect of Renewing the Operating License of a Nuclear Power Plant," to Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," of 10 CFR Part 51. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," issued May 1996, provides the technical basis for these issues. 10 CFR Part 51 and NUREG-1437 identified 92 environmental issues relevant to license renewals; of these, 69 were considered "generic" issues, or applicable to all nuclear power plants. The other 23 issues have required plant-specific reviews. The NRC reviews and reevaluates these issues, whether generic or plant-specific, for each license renewal application. For the plant-specific issues, the applications also must provide their own plant-specific analysis of the issue(s), which the NRC staff then independently reviews.

In addition, RG 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," Revision 2, issued July 1976, Supplement 1, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses," issued September 2000, provides guidance to applicants preparing environmental reports for license renewal. Supplement 1, "ESRP for Operating License Renewal," issued March 2000, to NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan [ESRP]," guides the NRC staff's review of the environmental issues associated with license renewal.

In consideration of lessons learned and knowledge gained from previous license renewal environmental reviews, the NRC is currently revising its 10 CFR Part 51 regulations and guidance. The proposed rulemaking, draft revised NUREG-1437, and guidance documents were issued for public comment in 2009. The final rule update was affirmed by the Commission in December 2012. As compared to the 92 issues under the 1996 rule, the rule update identifies 78 environmental issues, including 17 requiring a plant-specific analysis. The change consolidates similar generic and plant-specific issues while adding new issues. It has also been changed to reflect the June 8, 2012, ruling by the U.S. Court of Appeals for the District of Columbia Circuit vacating the NRC's waste confidence decision and rule. The final revised environmental protection regulation, NUREG-1437, and associated guidance documents are expected to be issued in mid-2013.

14.1.4.2 Experience

The NRC issued the first renewed licenses for the Calvert Cliffs Nuclear Power Plant and the Oconee Nuclear Station in 2000. As of December 2012, 73 reactors (including Kewaunee) have received renewed licenses. Fifteen of the 73 reactors have completed 40 years of operation and are operating in the extended period. Ten[5] more reactors will enter the period of extended operation in 2013. On the basis of industry statements, the NRC expects that essentially all remaining plants will apply for license renewal.

14.1.4.3 Operating Beyond 60 Years

The provisions of 10 CFR Part 54 do not preclude subsequent license renewals after the initial renewal. The earliest that a licensee can submit a license renewal application is 20 years before the expiration of its current license; therefore, a licensee is eligible to apply for a subsequent license renewal once it enters the initial period of extended operation (the 20-year renewal period beyond its initial 40-year license period). While several industry representatives have expressed an interest in license renewal beyond 60 years, the Commission has not received any formal letter of intent to pursue such a renewal.

To encourage early and proactive discussion of factors potentially affecting subsequent license renewal decisions and following the initial February 2008 workshop, the Commission and DOE jointly sponsored a second workshop on U.S. nuclear power plant life extension research and development on February 22 – 24, 2011. The Commission and DOE also co-hosted the third International Symposium on Nuclear Plant Life Management for Long Term Operation organized by the IAEA on May 14 – 18, 2012. In addition, the Commission is working with other national regulators and nongovernmental organizations to implement an International Forum for Reactor Aging Management that would create a network of international experts who would exchange information on operating experience, best practices, and emerging knowledge. Finally, the Commission began engaging the stakeholders through several public meetings and Webinars in May and November 2012 to solicit feedback from them on issues that need to be considered for operation beyond 60 years.

Based on the results of the workshops, symposium, and public interactions, major areas of research for a subsequent license renewal include: aging management of reactor vessel and internal materials, cable insulation, buried piping, submerged structures, and concrete exposed to high temperature and radiation. The industry is responsible for conducting the necessary research to provide the technical basis for operation beyond 60 years. The NRC continues to track industry's work in this area, evaluate areas for research, and gather data to help assess the effectiveness of licensee's aging management programs, and provide confirmatory research on the results of industry's work.

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5 On October 22, 2012, Dominion Resources, the operator of Kewaunee Power Station, announced that it would close the plant and move to safe shutdown in the second quarter of 2013, lowering the number of units entering the period of extended operation in 2013 from eleven to ten. The station will be under NRC oversight throughout the decommissioning process.

14.1.5 The United States and Periodic Safety Reviews

To a large extent, the international community conducts periodic safety reviews (typically carried out every 10 years) to assess the cumulative effects of plant aging, plant modifications, operating experience, technical developments, and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices, with the objective of ensuring a high level of safety throughout the plant's operating lifetime.

Some countries use routine comprehensive safety assessment programs that deal with specific safety issues, significant events, and changes in safety standards and practices as they arise. These programs, if applied with appropriate scope, frequency, depth, and rigor, achieve the same review standards and objectives as a periodic safety review. Some countries also use periodic safety reviews to support the decisionmaking process for long-term operation or license renewal. However, alternate processes, such as the NRC license renewal process, are considered equally adequate and acceptable.

This section explains how the U.S. regulatory approach provides a continuum of assessment and review that ensures public health and safety throughout the period of plant operation. Plant safety is maintained, and aspects are improved, by a combination of the ongoing NRC regulatory process, oversight of the current licensing basis, backfitting, broad-based evaluations, license renewal, and licensee initiatives that go beyond the regulations.

14.1.5.1 The NRC's Robust and Ongoing Regulatory Process and the Current Licensing Basis

Before issuing an operating license, the NRC determines that the design, construction, and proposed operation of the nuclear power plant satisfy the NRC's requirements and reasonably ensure the adequate protection of public health and safety. However, the licensing basis of a plant does not remain fixed for the 40-year term of the operating license. The licensing basis evolves throughout the term of the operating license because of the NRC's continuing regulatory activities and the licensee's activities.

The NRC carries out many regulatory activities that, when considered together, constitute a process providing ongoing assurance that the licensing bases of nuclear power plants provide an acceptable level of safety. This process includes inspections (both periodic regional inspections as well as daily oversight by the resident inspectors), audits, investigations, evaluations of operating experience, regulatory research, and regulatory actions to resolve identified issues. The NRC's activities may result in changes to the licensing basis for nuclear power plants through the issuance of new or revised regulations, orders, or confirmatory action letters. The agency also publishes the results of operating experience analysis, research, or other appropriate analyses through generic communication documents such as bulletins, INs, RISs, and Generic Letters (GLs). Licensee responses to these documents may also propose changes to the plant's licensing basis when appropriate. In this way, the NRC's consideration of new information gives ongoing assurance that the licensing basis for the design and operation of each nuclear power plant provides an acceptable level of safety. This process continues for plants that receive a renewed license to operate beyond the original operating license.

In addition to NRC-required changes in the licensing basis, a licensee may also voluntarily seek changes to the current licensing basis for its plant. These changes are subject to the NRC's formal regulatory controls on changes (such as those described in 10 CFR 50.54, "Conditions of Licenses," 10 CFR 50.59, 10 CFR 50.90, and 10 CFR 50.92). These regulatory controls ensure that licensee-initiated changes to the licensing basis are documented and that the licensee obtains NRC review and approval, if necessary, before implementing them. In accordance with 10 CFR 50.59(d)(2), the licensee must report to the NRC any changes or modifications it makes to the licensing basis without prior NRC review at least every 2 years. As stated in 10 CFR 50.71(e), the periodic update ensures the final safety analysis report contains the latest information. Region-based NRC inspectors perform a sampling inspection of those changes in accordance with the Reactor Oversight Process to ensure that the licensee has properly characterized the changes or modifications. The Reactor Oversight Process is the NRC's program to inspect, measure, and assess the safety performance of commercial nuclear power plants and to respond to any decline in performance. Annually, the Commission devotes a significant amount of resources to the oversight process. For example, each plant receives 6,000 to 10,000 hours of inspection every year. Additionally, more than 1,200 hours are spent evaluating licensing tasks at each plant. This level of effort gives the Commission the confidence that its oversight process produces a level of safety comparable to that of the periodic safety review process. Section 6.3.2 of this report provides a full description the NRC Reactor Oversight Process.

14.1.5.2 The Backfitting Process: Timely Imposition of New Requirements

In the late 1960s and again in the early 1980s, the NRC recognized the need for a process to determine when to address generic issues for all plants. The NRC decided to consider new requirements systematically rather than depending on other regulatory processes to decide on plant upgrades. As a result, the NRC developed the "backfitting" process and established the Committee to Review Generic Requirements to review NRC-staff-proposed backfits on licensees.

The Backfitting Rule, 10 CFR 50.109, "Backfitting," first issued in 1970 and substantially revised in 1985, applies to both generic and plant-specific backfits for power reactors. The rule defines a "backfit" as any modification of or addition to (1) facility systems, (2) facility structures, (3) facility components, (4) facility designs, (5) design approvals, (6) manufacturing licenses, or (7) procedures or organization required to design, construct, or operate a facility – any of which may result from the imposition of a new or amended rule or regulatory staff position.

In 1988, the NRC amended the Backfitting Rule to state that economic costs will not be considered in cases of backfits imposed to ensure, define, or redefine adequate protection of public health and safety, or common defense and security. Another exception to the need to prepare a backfit analysis is when backfits are imposed to ensure compliance with NRC requirements (i.e., an NRC license, regulation, or order), or conformance with written commitments by a licensee. These backfits are referred to as compliance and adequate protection backfits. For backfits other than compliance backfits, the NRC must determine that the proposed backfit will substantially increase the overall protection of public health and safety or the common defense and security and that the direct and indirect costs for the facility are justified in view of the increased level of protection.

Backfitting is permitted only after a formal, systematic review to ensure that changes are properly justified and suitably defined. The requirements of this process are intended to ensure order, discipline, and predictability and to optimize the use of NRC staff and licensee resources.

The controls on generic backfitting include a Committee to Review Generic Requirements review, which is a committee of senior managers from different NRC offices. Established in 1981, this committee operates under a charter that specifically identifies the documents to be reviewed and the analyses, justifications, and findings to be supplied to this committee by the NRC staff. Its objectives include eliminating unnecessary burdens on licensees, reducing radiation exposure to workers while implementing requirements, and optimizing use of NRC and licensee resources to ensure safe operation. Therefore, the Committee to Review Generic Requirements' charter is a key implementing procedure for generic backfitting, although the primary responsibility for proper backfit considerations belongs to the initiating organization.

14.1.5.3 The NRC's Extensive Experience with Broad-Based Evaluations

In the mid-1970s, the NRC recognized the importance of assessing the adequacy of the design and operation of currently licensed nuclear power plants, and understanding the safety significance of deviations from applicable current safety standards that may have been approved after those plants were licensed. It also recognized the importance of providing the capability to make integrated and balanced decisions about the need for backfit modifications at those plants.

Consequently, in 1977, the NRC initiated the Systematic Evaluation Program (SEP). From a list of approximately 800 potential issues and topics related to nuclear safety, the SEP found that the regulatory requirements for 137 issues had changed sufficiently to warrant evaluation. The staff compared the designs of 10 of the older plants to the licensing criteria delineated in the then-recently issued Standard Review Plan.[6] After further review, the staff determined that 27 issues required some corrective action at one or more plants and that resolution of those issues could lead to safety improvements at other operating plants built at about the same time. These 27 issues became known as the "27 SEP lessons learned."

In 1984, NRC staff presented the 27 SEP lessons learned to the Commission as part of a proposal for an integrated safety assessment program (ISAP). The staff developed this program to review safety issues for a specific plant in an integrated manner instead of continuing the SEP at other older operating reactors. In "Commission Policy Statement on the Systematic Safety Evaluation of Operating Nuclear Power Reactors," dated November 15, 1984, the Commission said that issues relating to the safety of operating nuclear power plants can be more effectively and efficiently implemented in an integrated, plant-specific review. For the first time, the Commission discussed probabilistic safety analysis as a method to obtain consistent and comparable results that could be used to enhance a safety assessment. The SEP process was transformed into the ISAP pilot program.

In May 1985, the NRC initiated the ISAP pilot at two plants, Millstone Unit 1 and Haddam Neck (Connecticut Yankee). The ISAP pilot identified some benefits; however, the Commission deferred extending it beyond the pilot phase until the staff gave an integrated package of options that clarified the relationship between the proposed follow-on program to the ISAP pilot (ISAP II) and the newly proposed individual plant examination process.

The Commission determined that, since ISAP II would be voluntary and the individual plant examination program, through the NRC's GL process, would require a licensee response, the staff should give priority to the individual plant examination program. Many of the same benefits that might have been derived through the proposed ISAP II were derived instead through the individual plant examination process (e.g., probabilistic safety analysis).

6 Standard review plans help ensure the quality and uniformity of staff reviews and provide a well-defined base from which to evaluate a licensee or applicant submittal. Standard review plans are also intended to make information about regulatory matters widely available, to enhance communication with interested members of the public and the nuclear power industry, and to improve the understanding of the staff review process.

In the late 1980s and throughout the 1990s, the NRC continued to strengthen its regulatory infrastructure and ensure the continued safe operation of commercial nuclear power plants through inspection, broad-based assessment, and, where appropriate, establishment of new generic requirements. For example, the Commission determined that licensees should assess the accessibility and adequacy of their design-basis information and determine whether their plants needed a design-basis reconstitution program. The Commission expressed its expectations in "Availability and Adequacy of Design Bases Information at Nuclear Power Plants; Policy Statement" in the *Federal Register* on August 10, 1992. The Commission also expanded the individual plant examination program to consider external events and, recognizing the relationship between maintenance, equipment reliability, plant risk, and safety, in 1991 the Commission issued the Maintenance Rule as codified in 10 CFR 50.65.

The Maintenance Rule requires licensees to monitor the performance or condition of SSCs against licensee-established goals continuously, to give reasonable assurance that these SSCs are capable of fulfilling their intended functions. The NRC verifies the licensee's implementation of the Maintenance Rule through the Reactor Oversight Process, periodic regional inspections, and daily oversight by the resident inspectors.

As late as 1991, some plants had not definitively resolved the 27 SEP lessons learned. As the staff considered a process to renew the operating licenses for the operating nuclear power plants, it assessed the best way to address these 27 issues.

Of the 27 issues, 4 had been completely resolved for all plants. One other issue was of such low safety significance that it required no additional action. The staff determined that none of the remaining 22 issues required immediate action to protect public health and safety. The staff placed these 22 issues into the established regulatory process for determining the safety significance of generic issues.[7]

14.1.5.4 License Renewal Confirms Safety of Plants

In developing the License Renewal Rule, the Commission concluded that issues material to the renewal of a nuclear power plant operating license are limited to those issues that the Commission determines are uniquely relevant to protecting public health and safety and preserving the common defense and security during the period of extended operation. Other issues would, by definition, be relevant to the safety and security of the public during current plant operation. Given the Commission's ongoing obligation to oversee the safety and security of operating reactors, the existing regulatory process within the present 40-year license term addresses issues related to current plant operation rather than deferring the issues until the time of license renewal. The NRC manages these issues by implementing the Reactor Oversight Process, generic communications, and the generic safety issues program.

7 A generic issue is a regulatory matter that existing regulations, guidance, or programs do not sufficiently address. Through its systematic assessment of plant operation, the NRC has identified certain issues that seem prevalent among plants. The NRC documents and tracks resolution of these "generic safety issues." The generic safety issue program provides for (1) identifying generic issues, (2) assigning them priorities, (3) developing detailed action plans for their resolution, (4) overseeing progress in their resolution by senior managers, and (5) informing the public of the status of progress in resolution. The resolution of these issues may involve new or revised rules, new or revised guidance, or revised interpretation of rules or guidance that affect nuclear power plant licensees or nuclear material certificate holders. The U.S. Congress requires the NRC to maintain this program.

The NRC issued the License Renewal Rule in 1995 (in 10 CFR Part 54). The license renewal process focuses on passive and long-lived SSCs because degradation in active components is more readily detected by complying with the Maintenance Rule. License renewal applicants are required to complete an environmental assessment and an integrated plant assessment[8] and to evaluate time-limited aging analyses. The current licensing basis must be maintained throughout the period of extended operation. Section 14.1.4 of this report describes the NRC license renewal process.

14.1.5.5 Risk-Informed Regulation and the Reactor Oversight Process

The NRC is actively increasing the use of risk insights and information in its regulatory decisionmaking. For reactors, risk-informed activities occur in the five broad categories of (1) applicable regulations, (2) licensing process, (3) Reactor Oversight Process, (4) regulatory guidance, and (5) risk analysis tools, methods, and data. Activities within these categories include revisions to technical requirements in the regulations; risk-informed technical specifications; a framework for inspection, assessment, and enforcement actions; guidance on risk-informed inservice inspections; and improved standardized plant analysis risk models.

In 2000, the NRC implemented a revised Reactor Oversight Process using risk insights and lessons learned from more than 30 years of regulating nuclear power plants. The previous oversight process evolved during a period when the nuclear power industry was less mature and there was much less operational experience on which to base rules and regulations. Therefore, very conservative judgments governed the rules and regulations. Significant plant operating events occurred with some frequency, and the oversight process tended to be reactive and prescriptive, closely observing plant performance for adherence to the regulations and responding to operational problems as they occurred.

After nearly four decades of operational experience and generally steady improvements in plant performance, the Reactor Oversight Process now focuses more of the agency's resources on the relatively small number of plants with performance problems. The process is a way to collect information about licensee performance, assess the information for its safety significance, and provide for appropriate licensee and NRC response, including corrective and enforcement actions, when appropriate.

The Reactor Oversight Process is a risk-informed tool that uses direct inspections and objective performance indicators to gauge and respond to plant performance. Together, the performance indicators and inspection findings give the information needed to support quarterly reviews of plant performance. The Reactor Oversight Process also features expanded semiannual reviews, which include inspection planning and a performance report (all posted on the NRC's public Web site). The Reactor Oversight Process is more effective at correcting performance or equipment problems today because the agency's response to problems is more focused and predictable. Section 6.3.2 of this report provides a full description of the NRC Reactor Oversight Process.

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8 An integrated plant assessment identifies and lists structures and components subject to an aging management review. These include "passive" structures and components that perform their intended function without moving parts or without a change in configuration or properties. Examples of these are the reactor vessel, the steam generators, piping, component supports, and seismic Category I structures. To be in scope, the item must also be long-lived to be considered during the license renewal process. Long-lived means the item is not subject to replacement based on a qualified life or specified time period.

14.1.5.6 Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations

As in many countries, U.S. nuclear power plant licensees are responsible for the safety of their facilities. This responsibility is embedded in their license and in the NRC's regulatory infrastructure. Under the regulatory umbrella, licensees routinely assess new technologies, off-normal conditions, operating experience, and industry trends to make informed decisions about safety enhancements to their facilities.

Under the U.S. regulatory structure, Appendix B to 10 CFR Part 50 requires that all nuclear power plant licensees maintain a quality assurance program. Quality assurance comprises all those planned and systematic actions necessary for adequate confidence that an SSC will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system that provide a way to control quality to predetermined requirements.

Licensees carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. Appropriately trained personnel who do not have direct responsibilities in the areas being audited perform these audits in accordance with written procedures or checklists. Management with responsibility in the area documents and reviews the audit results, and appropriate followup is initiated.

14.1.5.7 The NRC's Regulatory Process Compared with International Safety Reviews

IAEA and the Western European Nuclear Regulators' Association (WENRA) have developed guidance[9] and objectives for conducting periodic safety reviews that have much in common. Consistent with the IAEA guidance, periodic safety reviews are comprehensive assessments to determine:

- the adequacy and effectiveness of the arrangements and the SSCs (equipment) that are in place to ensure plant safety until the next periodic safety review or, where appropriate, until the end of planned operation (that is, if the nuclear power plant will cease operation before the next periodic safety review is due)
- the extent to which the plant conforms to current national and/or international safety standards and operating practices
- safety improvements and timescales for their implementation
- the extent to which the safety documentation, including the licensing basis, remains valid

For the reasons discussed above and summarized below, the shared objectives associated with the IAEA and WENRA periodic safety review guidance are substantively accomplished in the United States on an ongoing basis.

First, the NRC's regulatory process provides a robust foundation for ongoing assessments, evaluations, and, when appropriate, imposition of new requirements. Currently, the NRC and the U.S. nuclear industry consider new information in a more risk-informed manner as it becomes available; adjust the regulatory oversight and plant safety priority, respectively; and

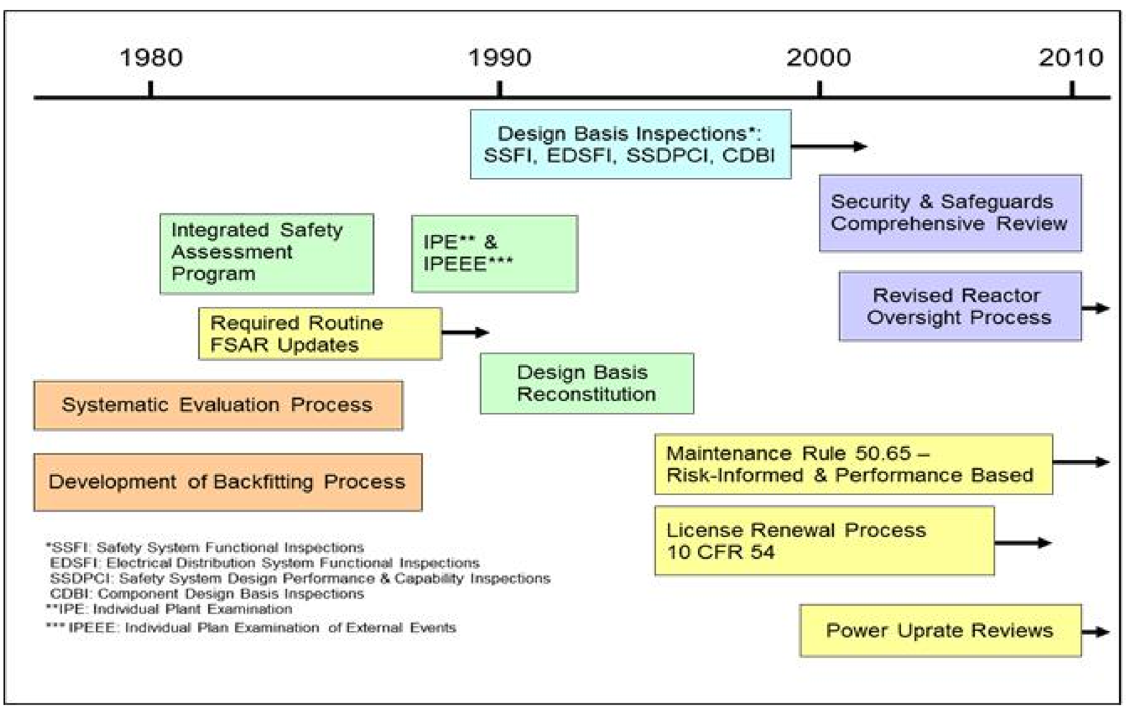
9 IAEA guidance appears in Specific Safety Guide SSG-25, "Periodic Safety Review of Nuclear Power Plants Safety," issued in 2013. WENRA has published several guidance documents on this subject. One of them is, "Position Paper on Periodic Safety Reviews (PSR) Taking into Account the Lessons Learnt from the TEPCO Fukushima Dai-ichi NPP Accident," WENRA Reactor Harmonization Working Group, dated March 2013.

provide ongoing assurance that the licensing basis for the design and operation of all nuclear power plants provides an acceptable level of safety. Development of the Maintenance Rule and License Renewal Rule are two examples of new requirements that serve this purpose.

Second, the NRC and the U.S. nuclear industry have a 30-plus-year history of implementing broad-based plant assessments. The regulatory history of implementing broad-based assessments is a direct result of an adaptive, probing, and independent regulatory process. These assessments have included the SEP, the ISAP, and the individual plant examinations. They provide additional confidence that plant safety continues to be the highest priority and that the NRC and industry continue to pursue enhancements that improve safety. As shown in the figure included below, over a period of almost 25 years, broad-based NRC assessments and regulatory initiatives have provided a continuum of assessment, improvement, and oversight, which ensures that licensed plants continue to operate safely.

The NRC's transition to a more risk-informed regulatory framework and the Reactor Oversight Process offers an ongoing approach and basis for implementing appropriate safety improvements, corrective actions, or process improvements, and provides confidence that the plant can continue to be operated safely. The NRC's more risk-informed approach helps ensure that resources are optimally focused on those issues most important to safety.

Finally, U.S. licensees establish performance expectations above the thresholds required by the NRC. These self-imposed expectations and initiatives -- over and above the regulations -- result from the licensee's self-described motivation to pursue excellence and by the recognition that safety and economics are directly linked in the competitive, free-market U.S. energy industry.



14.2 Verification by Analysis, Surveillance, Testing, and Inspection

Licensees are required to verify that they are operating their nuclear installations in accordance with the plant-specific design and requirements. The technical specifications and national consensus codes (for testing and periodic inspections) contain some of the requirements for verification.

In 10 CFR 50.55a, the NRC gives requirements for applying industry codes and standards to nuclear power reactors during design, construction, and operation. This section states, "Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b) through (g) of this section." In addition, 10 CFR 50.55a provides for alternatives to the ASME Code when authorized by the NRC.

Through analysis, surveillance, testing, and inspection, the NRC and licensees verify that the physical state and operation of nuclear installations continue to be in accordance with the designs, applicable national safety requirements, and operational limits and conditions. As discussed in Article 6 of this report, the NRC's Reactor Oversight Process includes inspections to verify that licensees are fulfilling their obligations to carry out such surveillances and testing and take corrective action.

Under special circumstances, the Commission may also require under 10 CFR 50.54(f) that licensees submit written statements for the Commission to determine whether the license should be modified, suspended, or revoked.

The NRC updates, revises, and improves existing regulatory programs in light of operating experience and significant new safety information. Article 19 of this report discusses these activities.

14.3 Fukushima Lessons Learned

As a result of the Fukushima accident, there were no changes to the existing Reactor Oversight Process or the NRC's processes for assessment and verification of safety. Immediately after the event, using the existing Reactor Oversight Process, the NRC conducted inspections, issued orders, INs, and bulletins to aid in determining the preparedness of U.S. nuclear power plants to withstand a similar event. These inspections, orders, INs, and bulletins are discussed in greater detail in Section 19.9. Further, the Reactor Oversight Process will be used to assess and verify that changes currently being implemented in response to Fukushima were completed properly.

ARTICLE 15. RADIATION PROTECTION

Each Contracting Party shall take the appropriate steps to ensure that, in all operational states, the radiation exposure to the workers and to the public caused by a nuclear installation shall be kept as low as reasonably achievable, and that no individual shall be exposed to radiation doses which exceed the prescribed national dose limits.

This section summarizes the authorities and principles of radiation protection, which include the regulatory framework, regulations, and radiation protection programs for controlling radiation exposure for occupational workers and members of the public. This section also discusses lessons learned from Fukushima. Article 17 of this report discusses radiological assessments that apply to licensing and facility changes.

15.1 Authorities and Principles

Generally, U.S. radiation control measures are founded on radiological risk assessments by the United Nations Scientific Committee on the Effects of Atomic Radiation and the U.S. National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation. The risk management recommendations that the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements issued reflect these assessments. On the basis of these assessments and recommendations, the U.S. EPA develops Federal guidance signed by the President of the United States, and "generally applicable radiation standards" for use by the other Federal agencies, including the NRC. The responsible agencies, such as the NRC, then establish regulations that consider these recommendations and standards. U.S. radiation protection programs are based on principles generally consistent with the principles espoused by ICRP: (1) it is known that large doses of ionizing radiation can be deleterious to human health, and (2) it is considered prudent to assume that small doses also may be harmful, with the probability of a deleterious effect being proportional to the dose. The U.S. programs acknowledge, include, and use the ICRP-recommended protection principles of "limitation," "justification," and "optimization" as appropriate.

Of these principles, "limitation" is the most practicable and most directly included in the regulatory structure. The regulations establish dose limits that cannot be exceeded without violating the regulations. There is a lengthy history of the doses being kept within the limits for workers ("Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 2011: Forty-Fourth Annual Report," NUREG-0713, Volume 33, issued April 2013) and members of the public living near nuclear power plants (NUREG/CR-2850, Volume 14, "Dose Commitments due to Radioactive Releases from Nuclear Power Plant Sites in 1992," issued March 1996.) More recent effluent release data are available on the NRC Web site at: http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

"Justification" is the recommendation that any activity involving radiation exposure be shown to be beneficial before the activity is undertaken. However, the risks or benefits of a new application of radioactive material can seldom be determined in advance with complete accuracy. Furthermore, radiation protection considerations are only one contributor to overall decisions on whether a particular exposure situation is justified. The "justification" activities in the United States are carried out during the licensing process. In general, the NRC will reject an application to use or produce radioactive materials if it determines that the application is not justified (i.e., that the overall benefit to society is outweighed by the risk of the radiation exposure associated with the activity). The licensing process under 10 CFR Part 50 does not directly address the justification for licensing a nuclear power plant. But, when a nuclear power plant is licensed, the environmental costs and benefits are evaluated in an environmental impact statement. This analysis is considered in the NRC's licensing decision.

Rather than using the term "optimization," the United States has used the term "as low as is reasonably achievable" (ALARA). In most circumstances, these two terms are consistent and represent the same underlying principle. As a guiding principle, ALARA (with varying terminology) dates back to 1939, in the United States and is defined in the regulations for occupational workers and members of the public.

For decades before 1994, 10 CFR Part 20, "Standards for Protection against Radiation," addressed the ALARA criterion for occupational radiation exposure, but more as an admonition than as a requirement. In 1994, the NRC changed the regulation to require that all licensees develop, document, and carry out an ALARA program. The NRC judges compliance with this requirement on the basis of a licensee's capability to track and, if necessary, reduce exposures, rather than on whether exposures and doses represented an absolute minimum or whether the licensee had used all possible methods to reduce exposures.

For control of radiation exposure from nuclear power plants to members of the public, the NRC modified 10 CFR Part 50 by adding Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." Issued in 1975, this appendix established design objectives to keep radioactive releases from nuclear power plants ALARA. The ALARA requirement led to the establishment of numerical objectives (for example, 0.00005 sieverts (Sv) (0.005 rem) in a year for the most highly exposed individual). Similar EPA requirements for other facilities soon followed. These NRC and EPA requirements are consistent with ICRP principles and result in public doses that are well below the local variation in doses from natural sources.

Although U.S. regulations generally are consistent with ICRP recommendations, certain considerations have limited the extent to which U.S. regulations match those of ICRP. One important consideration has been the U.S. desire for regulatory stability. Revising the regulations to incorporate every new ICRP position would impose a serious burden on the licensees without a commensurate benefit. Furthermore, for nuclear power reactors, new requirements are constrained by the Backfit Rule's requirements that any increase in regulatory requirements other than those required for compliance with existing regulations or the statutory standard of "adequate protection" be justified by a commensurate improvement in safety. Consequently, U.S. regulations were founded on older (rather than the most recent) ICRP recommendations. Nevertheless, the NRC directed the staff to work closely with ICRP and other national and international organizations to help develop revised recommendations. After publication of the new ICRP recommendations (ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," approved March 2007), the NRC staff initiated stakeholder dialogue on key issues, and provided options for Commission consideration in SECY-12-0064, "Recommendations for Policy and Technical Direction to Revise Radiation Protection Regulations and Guidance," dated April 25, 2012. In its SRM issued in December 2012, the Commission approved the staff continuing stakeholder dialogue and technical basis development to explore the benefits and effects of increasing alignment with ICRP. The Commission disapproved any change to the occupational limit for effective dose. The Commission approved the staff's development of the regulatory basis for a revision to 10 CFR Part 20 and parallel alignment of 10 CFR Part 50, Appendix I, to reflect the most recent methodology and use consistent terminology for dose assessment. Appropriate steps should be taken to assure that conforming changes are made as soon as practical to make these methods consistent throughout all NRC regulations. Additionally, the Commission directed the staff to work with stakeholders to explore alternatives for dealing with individuals who may be approaching the dose limits and improve reporting of occupational exposure by licensees. As part of this process, the NRC staff is currently in active dialogue with all segments of the licensed community in the United States.

15.2 Regulatory Framework

The NRC developed requirements for radiation protection to implement three laws that the U.S. Congress passed: the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974; and the Uranium Mill Tailings Radiation Control Act of 1978.

NRC regulations establish the primary direct controls over licensees. Various documents provide additional guidance and clarification, including RGs, topical reports (NUREG series), GLs, technical specifications, and license conditions. These documents are supported by international standards, consensus national standards, and authoritative recommendations (such as those of ICRP and the National Council on Radiation Protection and Measurements). However, these supporting documents have no official status unless they are referenced in or adopted by a regulation or documents providing regulatory guidance, such as RGs or Standard Review Plans. Of particular importance are NUREG-0800, which guides the staff in reviewing safety analysis reports, and RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued November 1978, which guides the applicant in writing safety analyses. Chapter 11, "Radioactive Waste Management," of NUREG-0800 addresses the control of radioactive effluents. Chapter 12, "Radiation Protection," addresses radiation protection. Chapter 15, "Transient and Accident Analysis," details how to calculate offsite and control room operator doses for design-basis accidents. Under 10 CFR 50.34(g), the facility must be evaluated against the standard review plan.

As Article 6 of this report discussed, the Reactor Oversight Process has cornerstones for radiation safety. The cornerstone for public radiation safety focuses on the effectiveness of the plant's programs in meeting applicable Federal limits on the exposure, or potential exposure, of members of the public to radiation and in ensuring that the effluent releases from the plant are ALARA. The cornerstone for occupational radiation safety focuses on the effectiveness of the plant's program(s) in maintaining the worker dose within the regulatory limits and providing occupational exposures that are ALARA.

15.3 Regulations

The regulations that apply to radiation protection are 10 CFR Part 20 and 10 CFR Part 50.

10 CFR Part 20. The NRC regulations in 10 CFR Part 20 establish requirements for radiation protection for all NRC licensees. The NRC gives additional requirements for specific operations and specific kinds of licenses in other parts of Title 10: Regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material;" 10 CFR Part 34, "Licenses for Industrial Radiography and Radiation Safety Requirements for Industrial Radiographic Operations;" 10 CFR Part 35, "Medical Use of Byproduct Material;" 10 CFR Part 39, "Licenses and Radiation Safety Requirements for Well Logging;" 10 CFR Part 40, "Domestic Licensing of Source Material;" 10 CFR Part 50; 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material;" 10 CFR Part 71, "Packaging and Transportation of Radioactive Material;" and

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."

The major revision of 10 CFR Part 20, issued in 1991, adopted the recommendations, quantities, and models recommended in ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection," issued January 1977, and in ICRP Publication 30, "Limits of Intakes of Radionuclides by Workers," dated 1978-1982, as well as some recommendations from National Council on Radiation Protection and Measurements Report No. 91, "Recommendations on Limits for Exposure to Ionizing Radiation," issued June 1987. The 1991 revision to 10 CFR Part 20 also adopted the same dose limit for a member of the public recommended in ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection," issued November 1990. The general requirements for radiation protection are provided in 10 CFR Part 20. This part is divided into subparts, with each subpart addressing a specific area of radiation protection, such as occupational and public dose limits, posting, surveys, monitoring, waste disposal, and reporting requirements.

The details of the requirements in 10 CFR Part 20 are not entirely consistent with international standards such as IAEA's General Safety Requirements Part 3 (GSR-3), "Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards – Interim Edition General Safety Requirements," issued November 2011. The main areas of difference between 10 CFR Part 20 and the IAEA Basic Safety Standards include: (1) the use of the effective dose equivalent in 10 CFR Part 20 versus the use of the effective dose in the IAEA standards, (2) an annual occupational dose limit on the effective dose equivalent of 0.05 Sv (5 rem) in 10 CFR Part 20 versus 0.02 Sv (2 rem) averaged over 5 years, with a maximum of 0.05 Sv (5 rem) in any year, in the IAEA standards, and (3) use of the biokinetic models from ICRP Publication 30 in 10 CFR Part 20 versus the more recent models used in the IAEA standards. NRC licensees are permitted to use the effective dose in place of the effective dose equivalent and to use the more recent internal dosimetry models in place of those recommended in ICRP Publication 30, with prior NRC approval.

In addition, many licensees and agencies have administrative dose limits similar to or lower than those in the IAEA Basic Safety Standards. Most other licensees operate at occupational doses far below those limits and standards and therefore are considered ALARA. In some cases, the occupational doses do exceed 0.02 Sv per year (2 rem per year), but these are a very small fraction of the total, and efforts are continuing to reduce these doses to lower levels. The current 10 CFR Part 20 provides a level of radiation protection that in almost all situations is comparable to that provided by international standards.

<u>10 CFR Part 50</u>. Although 10 CFR Part 50 is the principal regulation addressing the safety of nuclear power plants, only a small section of it directly addresses radiation protection. Even so, the sections of 10 CFR Part 50 that affect radiation protection are significant. Of particular importance are 10 CFR 50.34a, Appendix I, and 10 CFR 50.34(g), which require NRC review of the in-plant radiation protection program. In 10 CFR 50.36a, the NRC also requires licensees to limit effluents from nuclear power reactors to the values in Appendix I to 10 CFR Part 50. The revised dose criteria for design-basis accidents appear in 10 CFR 50.34(a)(1)(ii)(D) for licensing actions after implementation of the revised rule in 1997. (The dose criteria for siting and determining the exclusion area low population zone and population center distance for nuclear power reactors appear in 10 CFR 100.11(a).) The Commission has approved the staff recommendation to develop a regulatory basis for updating Appendix I to 10 CFR Part 50, to use the most recently recommended methodologies and terminology. The staff has initiated this activity.

15.4 Radiation Protection Activities

Radiation protection activities apply to occupational workers and to members of the public.

15.4.1 Control of Radiation Exposure of Occupational Workers

In addition to focusing on personnel qualifications for licensing, the NRC's oversight and regulation of radiation protection programs ensure that the safety analysis report and radiation protection plan properly address each item in 10 CFR Part 20, as well as the provisions for instructions to workers in 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations." Guidance is provided in relevant RGs, such as RG 1.8, and RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3, issued June 1978.

Once the NRC issues a license, it maintains an active regulatory program that includes routine inspection and monitoring of nuclear plants to alert NRC staff of potential problems in radiation safety. Significant health physics problems can trigger significant reactive regional inspections or a generic communication to the industry.

The NRC staff has been collecting the annual occupational exposure data for light-water reactors since 1969. Because the amount and kind of maintenance performed strongly influence the doses, the individual plant collective doses fluctuate from year to year. Still, clear trends are evident. Using the average collective dose per reactor as the reference, statistical analysis shows that the doses varied almost randomly before the accident at Three Mile Island Unit 2. Thereafter, the doses increased as a result of the extensive modifications required of all nuclear power plants in response to new NRC requirements. The average collective dose reached a peak of 7.91 person-Sv (791 person-rem) per reactor in 1980. Since then, doses have declined almost steadily to the current level below 1 person-Sv (100 person-rem) per reactor, where they have remained for the past 8 years (2004-2011, the last year for which data have been compiled). The 2011 average collective dose value of 0.84 person-Sv (84 person-rem) per reactor was the second lowest average collective dose recorded since data collection began in 1969 (the average collective dose for 2010 was 0.83 person-Sv (83 person-rem)). Although the average doses for both PWRs and BWRs have been steadily declining, the average BWR dose has exceeded the average PWR dose since 1974. Over the past 5 years, the average BWR dose has exceeded the average PWR dose by roughly a factor of 2 (in part because of the higher average dose rates and larger work force at BWRs).

In 2011, the 137,360 workers at nuclear plants received 87.71 person-Sv (8,771 person-rem) for an average of 0.00064 Sv (0.064 rem) per worker. This represents a 93-percent drop in average worker dose from the 1973 value of 0.0095 Sv (0.95 rem) per worker.

15.4.2 Control of Radiation Exposure of Members of the Public

The regulations in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," control radiation exposures to members of the public. In addition to the 1.0 millisievert (100 millirem) annual dose limit in 10 CFR Part 20, the EPA regulations in 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," establish a regulatory standard such that the annual dose to a member of the public from exposures to sources associated with the entire uranium fuel cycle does not exceed 0.25 millisievert (25 millirem).

The regulations in 10 CFR 50.34a, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, define the ALARA plant objectives for effluents. Appendix I also specifies effluent monitoring, environmental monitoring, investigations, land-use censuses, and reporting. Section IV.B of Appendix I to 10 CFR Part 50 requires the licensee to establish an appropriate surveillance and monitoring program that will accomplish the following:

- Provide data on quantities of radioactive material released in liquid and gaseous effluents.
- Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure.
- Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.

Appendix I requirements are supplemented by 10 CFR Part 20.1501, “General,” which requires, in part, that a licensee perform surveys to evaluate potential radiological hazards and to demonstrate compliance with the public dose limits in 10 CFR 20.1301 and 10 CFR 20.1302. Therefore, a licensee is responsible for performing radiation surveys at its facility for radioactive materials that have the potential to affect workers and members of the public. Potential survey sites can include areas that licensed radioactive material has previously affected, as well as areas that licensed radioactive material may affect in the future. For onsite spills and leaks that may contain licensed radioactive material, 10 CFR 20.1501 requires a licensee to perform appropriate radiation surveys and monitoring to determine the radiological hazard (i.e., dose assessment) to workers and to determine if there is a viable pathway to the unrestricted area that could result in a potential radiological hazard to members of the public. The surveys and monitoring can continue over a period of time or become an ongoing monitoring program so that the licensee can adequately characterize the extent and source of the contamination from the spills or leak.

Since 2004, there have been several discoveries of radioactive ground water contamination at nuclear power facilities in the United States. Investigation has determined that most of the contamination resulted from undetected leakage from facility SSCs that contained or transported radioactive liquids. All unmonitored releases resulted in varying levels of onsite tritium ground water contamination, with two facilities detecting low levels of tritium (below EPA drinking water standards) in offsite residential drinking wells. Current data show no immediate public health effects and a very low probability that there will be an effect in the future.

The NRC has responded to reports of ground water contamination by carrying out inspections, assessing the safety significance of these events, and evaluating licensee performance in finding and taking corrective actions. The NRC has also issued IN 2004-05, “Spent Fuel Pool Leakage to Onsite Groundwater,” dated March 3, 2004, and IN 2006-13, “Ground-Water Contamination due to Undetected Leakage of Radioactive Water,” dated July 10, 2006, describing unmonitored and unplanned leakage at several nuclear power stations.

Both the NRC and the nuclear industry have worked to resolve the technical and programmatic issues leading to the ground water contamination events. In March 2006, the NRC Executive Director for Operations established a Liquid Radioactive Release Lessons Learned Task Force to assess lessons learned from the unmonitored release of radioactive liquid to the environment at power reactor sites and to recommend possible agency actions. The task force completed its assessment and issued its report on September 1, 2006. The most significant conclusion was that these events had no public health effect. However, because of the high level of public concern and the potential for contaminated ground water to migrate off site undetected, the task force made several recommendations to the NRC. In response to the task force recommendations, the NRC revised its guidance in RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," and RG 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants," both issued June 2009, to clarify its expectations concerning monitoring and reporting of leaks and spills.

In parallel with the NRC's efforts, the nuclear industry also responded to the ground water contamination events. The NEI has developed a voluntary Groundwater Protection Initiative that licensees have endorsed unanimously. The initiative established that each participating nuclear plant to have a plan in place by July 2006, which established several short-term actions, such as developing an enhanced communications protocol to ensure notification of State and local officials of less significant unmonitored release events. The industry initiative also included several long-term actions to improve leak detection monitoring capability and understanding of site hydrology and geology.

The NRC has initiated a special inspection effort to monitor the licensee's implementation of the industry's Groundwater Protection Initiative. As a result of the enhanced monitoring, the NRC has identified several additional occurrences of low-level tritium contamination in onsite ground water. To date, levels of contamination have been below any NRC-required reporting level and well below the ALARA dose objectives in Appendix I to 10 CFR 50. However, the NRC continues to oversee licensee's responses to each of these occurrences.

The nuclear power industry also voluntarily developed the Underground Piping and Tanks Integrity Initiative with a goal to provide reasonable assurance of structural and leakage integrity of in-scope underground and buried piping and tanks. This initiative places special emphasis on components that contain licensed radioactive materials. The intent of the Underground Piping and Tanks Integrity Initiative is to:

- drive proactive assessment and management of the condition of piping and tanks that fall within the initiative's scope
- ensure sharing of industry experience
- drive technology development to improve upon available techniques for inspecting and analyzing underground piping and tanks

In addition, in March 2010, the NRC established a task force to evaluate its regulatory framework associated with ground water protection. The objective of the task force was to evaluate NRC actions to date addressing buried piping leaks and whether those actions needed to be augmented. The report "Groundwater Task Force Final Report," dated June 2010, documents the task force's observations, conclusions, and recommendations in a number of areas, including policy and communications.

On August 15, 2011, the Commission issued SRM-SECY-11-0019, "Senior Management Review of Overall Regulatory Approach to Groundwater Protection," which directed the staff to propose options for ground water protection. The options were provided on March 29, 2012, in SECY-12-0046, "Options for Revising the Regulatory Approach to Ground Water Protection." In the SRM for SECY-2012-0046, dated May 24, 2012, the Commission instructed the staff to continue the current regulatory approach to ground water protection including the recently imposed requirements contained in the decommissioning planning rule (i.e., to require licensees to minimize the introduction of residual radioactivity into the site and to perform subsurface (ground water) surveys.)

The NRC is also actively staying abreast of industry efforts through participation in ASME Boiler and Pressure Vessel Committees, interaction with EPRI personnel, information sharing with other agencies and participation in international meetings to discuss inspection technology for buried and underground piping. The industry is working to develop inspection technology for remote, nondestructive acquisition of structural integrity information in buried piping systems. Some new robotic remote-delivery technology has been deployed; other new technology is under development. The agency has established milestones in the "Buried Piping Action Plan," Revision 2, dated November 30, 2011, to periodically assess the performance of available technology and the need to make changes to the current regulatory framework.

Meanwhile, the NRC staff continues to provide the public with current information on ground water contamination beneath the nuclear plant sites on its public Web site. Information includes the annual radiological effluent reports for each nuclear site, the annual environmental monitoring report for each site, a radioactive effluent summary report by calendar years, and a list of the plant sites with licensed radioactive material in ground water.

15.5 Fukushima Lessons Learned

After the Fukushima accident, the NRC identified and developed action plans for the following issues related to radiation protection:

- emergency preparedness enhancements for prolonged station blackout (SBO) and multi-unit events
- evaluation of the Emergency Response Data System capability
- emergency preparedness decisionmaking, radiation monitoring, and public education
- evaluation of the emergency planning zone size
- evaluation of the prestaging of potassium iodide beyond 10 miles

In a post-Fukushima effort to better estimate source term, the NRC is conducting research to update RASCAL (Radiological Assessment System for Consequence Analysis) to better estimate source term considerations for protective actions. The next version of RASCAL will provide multi-unit assessment capability. In addition, as a result of insights from the Fukushima lessons learned, the NRC is beginning to use State-of-the-Art Reactor Consequence Analyses research to better inform source term options in RASCAL.

ARTICLE 16. EMERGENCY PREPAREDNESS

(i) Each Contracting Party shall take the appropriate steps to ensure that there are onsite and offsite emergency plans that are routinely tested for nuclear installations, and cover the activities to be carried out in the event of an emergency.

(ii) For any new nuclear installation, such plans shall be prepared and tested before it [the installation] commences operation above a low power level agreed [to] by the regulatory body.

(iii) Each Contracting Party shall take appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.

(iv) Contracting Parties that do not have a nuclear installation on their territory, insofar as they are likely to be affected in the event of a radiological emergency at a nuclear installation in the vicinity, shall take the appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.

This section discusses (1) the background of emergency planning in the United States (2) offsite emergency planning and preparedness, (3) emergency classification system and emergency action levels, (4) recommendations for protective action in severe accidents, (5) inspection practices and regulatory oversight, (6) response to an emergency, (7) communications with neighboring states and international arrangements, (8) communications with the public, and (9) lessons learned from the Fukushima event.

16.1 Background

The NRC's responsibilities for radiological emergency preparedness stem from the agency's licensing functions under the Atomic Energy Act and the Energy Reorganization Act. Both statutes authorize the Commission to issue regulations that it deems necessary to fulfill its responsibilities under the acts. Following the accident at Three Mile Island Unit 2 in March 1979, the NRC amended the regulations to require significant changes in emergency planning and preparedness for U.S. commercial nuclear power plants.

The NRC's emergency planning regulations are an important part of the regulatory framework for protecting public health and safety and have been adopted as an added conservatism in the NRC's defense-in-depth safety philosophy of multiple-barrier containment and redundant safety systems. Before a full-power operating license can be issued, NRC regulations require a finding that there is reasonable assurance that adequate measures to protect public health and safety can and will be taken in a radiological emergency (10 CFR 50.47(a)).

Emergency planning in the United States recognizes that a spectrum of accidents could exceed the design-basis accidents that nuclear plants are required to accommodate without significant public health and safety effects. For design-basis accidents, the small releases that might occur would not likely require responses such as evacuating or sheltering the general public.

These actions become important only when considering accidents that are much less probable than design-basis accidents. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," issued December 1978, and NUREG-0654/FEMA-REP-1 (NUREG-0654), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, issued November 1980, describe the emergency planning basis. NUREG-0654 is being revised to align with the NRC emergency preparedness rule changes, which became effective in December 2011, and with the revised FEMA Radiological Emergency Preparedness Program manual issued in 2011.

16.2 Offsite Emergency Planning and Preparedness

The accident at Three Mile Island Unit 2 revealed that better coordination and more comprehensive emergency plans and procedures were needed if the NRC and the public were to have confidence in the readiness of onsite and offsite emergency response organizations to respond to a nuclear emergency. Before the accident at Three Mile Island Unit 2, there was no clear obligation for State and local governments to develop emergency plans for radiological accidents, and the Federal role was one of assistance and guidance. After the accident, the NRC amended its emergency planning regulations to require, as a condition of licensing, that each applicant or licensee submit the radiological emergency response plans of the State, Tribal and local governments that are within the plume exposure zone, as well as the plans of State governments within the ingestion pathway zone (10 CFR 50.33(g) and 10 CFR 50.54(s)).

In December 1979, the President directed FEMA to take the lead in ensuring the development of acceptable State, Tribal, and local offsite emergency plans and activities for nuclear power plants. The NRC and FEMA regulations, as well as a memorandum of understanding between the two agencies, contained in Appendix A "Memorandum of Understanding between Federal Emergency Management Agency and Nuclear Regulatory Commission," to 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness," dated June 17, 1993, subsequently codified FEMA's role and responsibilities.

FEMA provides its findings on the acceptability of the offsite emergency plans and preparedness to the NRC, which has the ultimate responsibility for determining the overall acceptability of radiological emergency plans and preparedness for a nuclear power reactor. The NRC will not issue a license to operate a nuclear power reactor unless it finds that the condition of onsite and offsite emergency preparedness provides reasonable assurance that protective measures can and will be taken in a radiological emergency. The NRC bases its decision on a review of the FEMA findings and determinations on whether State and local emergency plans are adequate and can be carried out, and on its own assessment of whether the onsite emergency plans are adequate and can be implemented (10 CFR 50.47(a)).

The principal guidance for preparing and evaluating radiological emergency plans for licensee, State, and local government emergency planners is NUREG-0654/FEMA-REP-1, a joint NRC and FEMA document. NUREG-0654/FEMA-REP-1 gives evaluation criteria for an acceptable way to meet the emergency planning standards in the NRC and FEMA regulations (10 CFR 50.47(b) and 44 CFR Part 350, respectively). These criteria provide a basis for licensees, States, Tribal, and local governments to develop acceptable emergency plans.

The NRC and FEMA coordinate their evaluation of periodic emergency response exercises and require all operating nuclear power plant sites to conduct an exercise every 2 years, as outlined in Section IV.F.2(b) of Appendix E to 10 CFR Part 50. These mandatory full-participation exercises are integrated efforts by the licensee, State, Tribal, and local radiological emergency response organizations that have a role in support of the licensee's emergency plan. The NRC evaluates the licensee's performance, while FEMA evaluates State, Tribal, and local agencies' responses. In some cases, other Federal response agencies also participate in these exercises. Any weaknesses or deficiencies that the NRC or FEMA identify as a result of the exercise must be corrected through appropriate remedial actions. Section IV.F.2(d) of Appendix E to 10 CFR Part 50, requires the offsite response agencies to participate in biennial exercises of their plume exposure pathway plans every 2 years, and for the State to participate in an ingestion pathway exercise with a nuclear power plant located within its State every 6 years. However, there are no requirements to involve members of the public in any of the emergency preparedness exercises.

Through the Steering Committee for Emergency Planning, established under the FEMA and NRC memorandum of understanding, both agencies discuss and coordinate on the interpretation and implementation of existing regulations and guidance; the consistent evaluation of each respective agency's radiological emergency preparedness programs and resolution of identified deficiencies; and the development and implementation of proposed changes to radiological emergency preparedness-related regulations and guidance.

16.3 Emergency Classification System and Emergency Action Levels

A licensee or applicant at a U.S. nuclear power plant is required to develop a standard emergency classification and action level scheme based on facility system and effluent parameters (10 CFR 50.47(b)(2)). Appendix E (Section IV.C.1) of 10 CFR Part 50 defines four emergency classification levels in order of increasing severity: (1) notification of an unusual event, (2) alert, (3) site area emergency, and (4) general emergency. The specific class of emergency is declared on the basis of plant conditions that trigger the emergency action levels.

Licensees, and State, Tribal, and local agencies have established specific procedures for carrying out emergency plans for each emergency classification level. The event classification, declared by the licensee, initiates appropriate actions for that class, including notification of offsite authorities, activation of onsite and offsite emergency response organizations, and, where appropriate, protective action recommendations for the public.

NUREG-0654/FEMA-REP-1 defines and gives examples of initiating conditions for the four emergency classifications. These conditions form the basis for each licensee to establish specific thresholds and indicators, known as "emergency action levels." Emergency action levels reflect specific plant conditions (e.g., plant system status, in-plant and effluent radiological parameters, and other in-plant hazards) or external events (e.g., flooding, earthquakes, high winds) for each of the four emergency classifications.

In RG 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 4, issued July 2003, the NRC endorsed the guidance in NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," Revision 2, issued January 1992; and NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, issued January 2003, as acceptable methods for developing emergency action levels. In July 2009, the NRC subsequently endorsed NEI-07-01, "Methodology for Development of Emergency Action Levels, Advanced Passive Light Water Reactors," Revision 0, which provided more specific guidance in support of new reactor license applications.

While not required under existing U.S. emergency preparedness regulations contained in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, a number of procedures guide onsite licensed reactor operator actions depending on the nature and extent of events at the plant. These events, such as a loss of offsite electrical power, are within the plant's design-basis and addressed by various plant procedures, typically abnormal operating procedures, alarm response procedures, and emergency operations procedures. These procedures instruct the plant operators on the steps necessary to take the plant from full-power operation to a safe shutdown condition, if necessary, based on the severity of the event. Emergency operating procedures have long been part of the NRC's safety requirements. Numerous regulatory guides and technical reports address the development of emergency operating procedures and their use (e.g., NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident" and NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980).

The nuclear industry developed severe accident management guidelines (SAMGs) in response to the TMI accident based on extensive research on severe accident phenomena. Their purpose is to enhance the ability of plant operators to manage accident sequences that progress beyond emergency operating procedures and other applicable plant procedures. While not required under U.S. emergency preparedness regulations, SAMGs are intended for use by plant technical staff, usually located in emergency support facilities activated under the emergency plan, in support of onshift control room operators. In GL 1988-20, , "Accident Management Strategies for Consideration in the Individual Plant Examination Process," Supplement 2, dated April 4, 1990, the NRC encouraged, but did not require, licenses to develop and implement SAMGs. Since SAMGs are voluntary, formal training and licensing of plant operators and emergency preparedness regulations do not address them. Currently, Recommendation 8 from the NRC's NTTF review of insights from the Fukushima accident addresses the integration of emergency operating procedures and SAMGs, along with extensive damage mitigation guidelines following the terrorist events of September 11, 2001, to specify clear command and control strategies for implementation and appropriate qualification and training for decisionmakers during emergencies.

16.4 Recommendations for Protective Action in Severe Accidents

The technical basis and guidance for developing protective action strategies for use during a nuclear power plant event resulting in a general emergency classification in the United States appear in NUREG-0654/FEMA-REP-1, Revision 1, Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents," issued November 2011, and EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992, which is currently under revision. NUREG-0654/FEMA-REP-1, Supplement 3, "Guidance for Protective Action Strategies," reflects the conclusions developed from analysis of a spectrum of nuclear power plant core melt accident scenarios. These analyses are documented in NUREG/CR-6953, "Review of NUREG-0654, Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents," Volumes 1, 2, and 3.

Although a general emergency is a serious event and warrants protective action, it is not necessarily synonymous with a "severe accident" as that term is used in U.S. nuclear power plant accident analyses. NUREG-0654/FEMA-REP-1, Supplement 3, recognizes the disparity between a severe accident with early release and other general emergency conditions, and provides scenario-specific protective action decision guidance. Additionally, it provides guidance for the consideration of evacuation time estimates and for the immediate evacuation of those closest to the nuclear power plant and criteria for the expansion of initial protective actions.

The agency provides guidance for response procedures and training manuals for NRC staff in NUREG/BR-0150, Volume 1, Revision 4, “Response Technical Manual 96,” dated March 1996. The NRC considers evacuation and sheltering to be the two primary protective actions.

A supplemental protective action for the general population is using the thyroid-blocking agent potassium iodide. In 2001, the NRC amended its regulations for emergency planning associated with potassium iodide, 10 CFR 50.47(b)(10). This amendment requires that each State consider giving potassium iodide to the general public as a protective measure, supplementing the evacuation and sheltering protective actions. The NRC found that potassium iodide is a reasonable, prudent, and inexpensive supplement to evacuation and sheltering for specific local conditions. In January 2002, the NRC, in cooperation with the cognizant agencies, updated the Federal policy statement on potassium iodide prophylaxis to reflect the changes in NRC regulations.

The agency provides guidance for response procedures and training manuals for NRC staff in NUREG/BR-0150, “Response Technical Manual 96,” Volume 1, Revision 4, issued March 1996. The NRC’s guidance on evacuation and sheltering in the event of a nuclear power plant accident is consistent with guidance in IAEA TECDOC-953, “Method for the Development of Emergency Response Preparedness for Nuclear or Radiological Accidents,” and IAEA TECDOC-955, “Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident,” both issued in 1997.

16.5 Inspection Practices - Reactor Oversight Process for Emergency Preparedness

The NRC’s Reactor Oversight Process addresses emergency preparedness. The process allows licensees to manage their own emergency preparedness programs, including corrective actions, as long as the performance indicators and inspection findings are within an acceptable performance band. The NRC handles inspection findings through its significance determination process. Article 6 of this report discusses the NRC’s Reactor Oversight Process and significance determination process.

Emergency preparedness is one of the Reactor Oversight Process’ seven cornerstones of safety. The objective of this cornerstone is to “ensure that the licensee is capable of implementing adequate measures to protect the public health and safety during a radiological emergency.” Oversight of this cornerstone is achieved through three performance indicators and a supporting risk-informed inspection program. The performance indicators are drill and exercise performance, emergency response organization drill participation, and alert and notification system reliability. The performance indicator for drill and exercise performance monitors timely and accurate licensee performance in drills, exercises, and actual events when presented with opportunities to classify emergencies, notify offsite authorities, and recommend protective actions. The indicator for emergency response organization drill participation measures the percentage of key members of the licensee’s emergency response organization who have participated in proficiency-enhancing drills, exercises, training opportunities, or an actual event over a determinant amount of time. The alert and notification system reliability indicator monitors the reliability of the offsite alert and notification system, which is a critical link for communicating with the public.

The emergency preparedness cornerstone of the Reactor Oversight Process includes the following areas for inspection:

- Maintenance of Emergency Preparedness Program - Inspectors evaluate the licensees' efforts to identify and resolve program weaknesses, adequacy of internal program assessment activities, emergency plan change process, maintenance of equipment important to emergency preparedness, evacuation time estimate population monitoring, and implementation of emergency response facility maintenance.

- Drill Evaluation - Inspectors evaluate drills and simulator-based training evolutions in which shift operating crews and licensee emergency response organization members participate.

- Exercise Evaluation - Inspectors independently observe the licensee's performance in classifying, notifying, and developing recommendations for protective actions, and other activities during the exercise. Evaluated exercise scenarios are varied over an 8-year exercise cycle to include a hostile action event, no radiological release or minimal release not requiring public protective actions, and a rapidly progressing event. The inspectors assess whether the licensee's self-critique is consistent with their observations. The emergency preparedness performance indicators for drill and exercise performance rely upon the accurate determination of successful performance and the correction of identified weaknesses during the conduct of drills and exercises. If a licensee either fails to properly critique performance or correct identified weaknesses, then the validity of the drill and exercise performance indicators come into question. Performance problems with classification, notification, dose assessment and protective action recommendations are the highest priority inspection areas. Exercise evaluation results are provided in inspection reports available on the NRC public Web site. These inspection reports identify findings associated with a licensee's failure to either properly critique or correct weaknesses observed during the conduct of a licensee's drill and exercise program.

- Alert and Notification System Evaluation - Inspectors verify how well the testing program complies with program procedures.

- Emergency Action Level and Emergency Plan Changes - Inspectors review all of the licensee's changes to emergency action levels and a sample of changes to the emergency plan to determine if any of the changes have decreased the effectiveness of the emergency plan.

- Emergency Response Organization Staffing and Augmentation System - Inspectors review the augmentation system to determine whether, as designed, it will support augmentation of the emergency response organization in accordance with the goals for activating the emergency response facility.

- Reactor Safety/Emergency Preparedness - Inspectors verify that the data reported for the performance indicator values are valid.

It is important to note, however, that even though FEMA has no direct regulatory authority over State or local governments and their full-participation exercise evaluations are not considered inspections, FEMA's exercise findings carry substantial weight in the NRC regulatory process. FEMA notifies the State Government and the NRC of any significant deficiencies in offsite performance shortly after the exercise. FEMA also issues a formal exercise report within 90 days of the exercise's completion describing the FEMA exercise findings. Because of the potential effect of deficiencies on offsite emergency preparedness, findings are expected to be corrected within 120 days of the exercise. Failure of offsite organizations to correct deficiencies promptly could lead FEMA to withdraw its finding of "reasonable assurance." This would cause the NRC to assess the continued operation of the facility.

16.6 Responding to an Emergency

Fundamental changes in the response to national emergencies have occurred as a result of the publication of the National Response Framework in January 2008 and the update of its associated annexes. Additionally, DHS has revised and republished the National Incident Management System (NIMS) document in December 2008.

This section explains the roles of the NRC, other Federal agencies, licensees, States, and local governments during the response to an incident. It also explains the security issues associated with supporting the response efforts.

16.6.1 Federal Response

The Federal response structure has been revamped with the creation of DHS and the implementation of Homeland Security Presidential Directive 5, "Management of Domestic Incidents," dated March 4, 2003. This directive establishes the Secretary of Homeland Security as the primary Federal official for managing domestic incidents. Under the Homeland Security Act of 2002, DHS is responsible for coordinating Federal operations within the United States to prepare for, respond to, and recover from terrorist attacks, major disasters, and other emergencies.

DHS will assume overall Federal incident management coordination responsibilities when any one of the following four conditions applies:

(1) A Federal department or agency acting under its own authority has requested DHS assistance.

(2) The resources of State and local authorities are overwhelmed, and the appropriate State and local authorities have requested Federal assistance.

(3) More than one Federal department or agency has become substantially involved in responding to the incident.

(4) The President of the United States has directed the Secretary to assume incident management responsibilities.

In 2008, the governing documents outlining the responsibilities of the Secretary of Homeland Security, DHS, and other Federal, State, and local entities were updated. These documents were related to NIMS and the National Response Framework and its associated annexes.

NIMS is a comprehensive, national approach to incident management that is applicable at all jurisdictional levels and across functional disciplines. NIMS enables Federal, State, and local entities to work together to prevent, protect against, respond to, recover from, and mitigate the effects of incidents, regardless of cause, size, location, or complexity, to reduce the loss of life and property and harm to the environment. NIMS provides an organized set of scalable and standardized operational structures that is critical for allowing various organizations and agencies to work together in a predictable, coordinated manner.

NIMS works hand-in-hand with the National Response Framework. NIMS provides the template for the management of incidents, while the National Response Framework describes the structures and mechanisms for national-level policy for incident management. The National Response Framework provides guidance on Federal coordinating structures and processes to prepare for, respond to, and recover from domestic incidents such as terrorist attacks, major disasters, and other emergencies.

The Federal response to a potential nuclear or radiological incident is designed to support the efforts of the facility operator and offsite officials. For such emergencies, Federal response activities are carried out in accordance with the National Response Framework's Nuclear/Radiological Incident Annex, which describes the roles of DHS, coordinating agencies (e.g., the NRC during an incident with one of its licensees), and other supporting Federal agencies. During an incident that meets the criteria of Homeland Security Presidential Directive 5 (invoked during a terrorist-related incident or at a general emergency level for an NRC licensee), DHS is responsible for the overall domestic incident management, while the coordinating agency coordinates the Federal on-scene actions and helps State and local governments determine measures to protect life, property, and the environment. The coordinating agency may respond as part of the Federal response as requested by DHS under the framework, or in accordance with its own authorities. During less severe incidents, coordinating agencies will oversee the onsite response, monitor and support owner or operator activities (when there is an owner or operator), provide technical support to the owner or operator if asked, serve as the principal Federal source of information about onsite conditions, and, if asked, advise the State and local government agencies on implementing protective actions. The coordinating agency also will provide a hazard assessment of onsite conditions that might have significant offsite effects and ensure that onsite measures are taken to mitigate offsite consequences.

16.6.2 Licensee, State, and Local Response

The NRC recognizes the nuclear power plant operator (licensee) and the State or local government as the two primary decisionmakers during a radiological incident at a licensed power reactor. The licensee is primarily responsible for mitigating the consequences of an incident on site and recommending timely protective actions to State and local authorities. The States or local governments are ultimately responsible for implementing appropriate protective actions for public health and safety.

16.6.3 The NRC's Response

In fulfilling its legislative mandate to protect the public health and safety, the NRC has developed a plan and procedures detailing its response to incidents involving licensed material and activities (NUREG-0728, "NRC Incident Response Plan," Revision 4, issued April 14, 2005). In accordance with that plan, the NRC will initially assess any reported event and decide whether or how it will respond as an agency. To meet its statutory and regulatory obligations, the NRC will usually dispatch a team to the site for all serious incidents. The team may help the State interpret and analyze technical information, update other responding Federal agencies on event conditions, and coordinate any multi-agency Federal response.

Once the NRC has decided to respond as an agency, it activates the NRC headquarters Operations Center near Washington, DC, and the associated regional incident response center. The NRC headquarters Operations Center will then take the following actions: (1) maintain continuous communications with the facility, (2) assess the incident, (3) advise the facility operator and offsite officials, (4) coordinate the Federal radiological response with other Federal agencies, and (5) respond to inquiries from the national media. The staff at the NRC headquarters Operations Center includes emergency preparedness and response experts and personnel experienced with liaison activities. Because regional office personnel usually have firsthand knowledge of the details of the affected facility, early in an incident the Regional Administrator provides operational authority from the affected regional office and, if necessary, from the regional incident response center. When NRC onsite presence is required, the agency will dispatch a team from the affected regional office.

As soon as the NRC site team arrives at the facility and is ready to assume the agency's leadership role, it may be delegated certain responsibilities that may include the authority to direct the agency's onsite response.

The NRC site team consists of many technical specialists and representatives who respond to the designated response centers that the facility and offsite officials use to coordinate the response. These response centers include the affected State's emergency operations center, the first-responder's incident command post, the joint information center, established by the facility or local government to interact with the media, and, if necessary, the joint field office (the primary Federal incident management field structure that is usually established 48 to 72 hours after an incident). Through participation in these response centers, the NRC site team has access to wide-ranging State and Federal response assets, as well as to extensive radiological monitoring capabilities through DOE (i.e., field teams and aerial monitoring).

The NRC regularly participates in nuclear power plant and Federal interagency exercises each year to ensure its readiness to respond. The NRC also participates in the planning and conduct of the annual continuity of operations exercise and National Level Exercises each year. The NRC's participation in such exercises gives the agency a valuable perspective on multi-event response. This perspective improves interagency cooperation and imparts a better understanding of response roles during emergencies.

16.6.4 Aspects of Security that Support Response

Before September 11, 2001, security measures at nuclear facilities provided reasonable assurance that public health and safety would be protected in the event of an attack encompassed by the design-basis threat for radiological sabotage, which is described in 10 CFR 73.1, "Purpose and Scope." Following the events of September 11, 2001, the nuclear industry has significantly enhanced its defensive capability through the voluntary actions licensees have taken in response to NRC advisories and as required by the security orders issued in 2002 and 2003. These enhancements included a revised design-basis threat for radiological sabotage and security measures against threats from an insider, waterborne attack, vehicle bomb attack, and land-based assault. The NRC subsequently issued its revised design-basis threat regulations on March 19, 2007, and revised power reactor security regulations on March 27, 2009. These updated regulations incorporated provisions from these security orders and lessons learned during the implementation of these orders. The NRC will consider additional measures in the future, as necessary.

The NRC receives a substantial and steady flow of information from the national intelligence community, law enforcement, and licensees and continually evaluates this information to assess threats to regulated facilities or activities. The NRC works with a variety of other Federal agencies, particularly DHS and the Federal Bureau of Investigation, to ensure that security around nuclear power plants is well coordinated and that law enforcement responders are prepared for a significant event. If an event were to occur, the NRC would have significant resources accessible to it and as many as 18 Federal agencies available to help mitigate the radiological consequences of a serious accident or successful attack.

16.7 <u>Communications with Neighboring States and International Arrangements</u>

The NRC has agreements with its neighbors, principally Canada and Mexico, and commitments to IAEA. The NRC's bilateral arrangements with non-neighboring countries also address and promote sharing of information on emergency preparedness and resources.

Since 2001, the United States has participated fully in the International Nuclear Event Scale by evaluating operating reactor events and reporting to IAEA any events resulting in a categorization of International Nuclear Event Scale Level 2 or higher. The United States has also played a significant role on the IAEA's International Nuclear and Radiological Event Scale Advisory Committee, including supporting the negotiations that resulted in the expanded use of the International Nuclear and Radiological Event Scale for rating radiation and transport events. The NRC participates in the IAEA's Unified System for Information Exchange for Incidents and Events as the method for rapidly sharing nuclear or radiological event information with IAEA and its member countries.

Under its signed agreements with Canada and Mexico, the NRC will promptly notify and exchange information in the event of an emergency that has the potential for transboundary effects.Under its signed agreements with Canada and Mexico, the NRC will promptly notify and exchange information in the event of an emergency that has the potential for trans-boundary effects. The agreement with Canada, "Agreement between the Government of the United States of America and the Government of Canada on Cooperation in Comprehensive Civil Emergency Planning and Management," is implemented by the procedure specified in "Administrative Arrangement between the United States Nuclear Regulatory Commission and the Atomic Energy Control Board of Canada for Cooperation and the Exchange of Information in Nuclear Regulatory Matters." The agreement between the NRC and the Canadian Nuclear Safety Commission, which replaced the Atomic Energy Control Board, was most recently renewed in 2012.

The agreement with Mexico, "Agreement for the Exchange of Information and Cooperation in Nuclear Safety Matters," is implemented by the "Implementing Procedure for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters between the Nuclear Regulatory Commission of the United States of America and the Comision Nacional de Seguridad Nuclear y Salvaguardias of Mexico," both dated October 6, 1989. This agreement was most recently renewed in 2012.

Since both bilateral agreements' most recent renewals occurred after the Fukushima accident, the NRC and its Canadian and Mexican counterparts have placed increased focus on their commitment to share information not only in the event of an accident, but on a regular basis as part of an effort to enhance their respective emergency preparedness programs. The NRC and the Canadian Nuclear Safety Commission held two technical bilateral meetings in 2012 and have additional exchanges planned for 2013 to discuss emergency preparedness-related topics. The NRC held a planning meeting with the Comision Nacional de Seguridad Nuclear y

Salvaguardias in March 2012 to discuss future cooperation meetings and technical bilateral exchanges.

To meet the U.S. commitment under the IAEA Convention on Early Notification of a Nuclear Accident, the NRC will promptly notify IAEA if a serious accident occurs at a commercial nuclear power plant. Afterward, the NRC will work with the U.S. Department of State to update IAEA. The NRC also routinely communicates with the IAEA and its Canadian and Mexican counterparts during its emergency drills. Since 2001, the United States has fully participated in the International Nuclear Event Scale by evaluating operating reactor events and reporting to IAEA any events resulting in a categorization of International Nuclear Event Scale Level 2 or higher. The NRC actively participates in the IAEA's Unified System for Information Exchange for Incidents and Events (USIE) as the method for rapidly sharing nuclear or radiological event information with IAEA and its member countries.

16.8 Communications with the Public

One of the emergency planning standards for U.S. nuclear power reactors requires that information be made periodically available to the public on how they would be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), that the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) be established in advance, and that procedures have been established for coordinated dissemination of information to the public. If an emergency were declared, another emergency planning standard requires that the content of initial and followup public messages has been established; and that a means has been established to provide early notification and clear instruction to the population within the plume exposure pathway emergency planning zone. NUREG-0654/FEMA-REP-1 outlines the evaluation criteria for complying with the requirements of these emergency planning standards.

Appendix E (Section IV.D) to 10 CFR Part 50 describes licensee requirements for the prompt notification of the public in the event of a declared emergency and for the yearly dissemination of basic emergency planning information to the public located within the plume exposure pathway emergency planning zone, such as:

- the methods and times required for public notification and the planned protective actions if an accident were to occur
- general information on the nature and effects of radiation
- a listing of local broadcast stations that would be used to disseminate information during an emergency
- the use of signs or other measures to disseminate appropriate information to transient populations in the event of an accident

The NRC performs continuous outreach with licensees and respective State, Tribal, and local emergency response organizations to facilitate stakeholder interface and involvement on existing and proposed radiological emergency preparedness activities. The NRC outreach effort consists of: (1) attending nuclear industry and radiological emergency preparedness-related conferences and forums, (2) conducting public meetings on proposed changes to radiological emergency preparedness-related regulations and guidance, and (3) using the NRC Web site, blog posts, and periodic newsletters for outreach.

16.9 Fukushima Lessons Learned

Following the Fukushima event, the NRC undertook actions to enhance emergency preparedness with respect to communications and staffing given a multi-unit event and a prolonged SBO. The accident highlighted the need to identify the staff needed to respond to a multi-unit event given a prolonged SBO. In addition, the accident highlighted that communication equipment relied upon during an emergency must be operable to coordinate the event response during a prolonged SBO.

On March 12, 2012, the NRC issued an RFI to all power reactor licensees and holders of construction permits to obtain information that would help the staff to evaluate the NTTF Recommendation 9.3 as discussed in NUREG-1650, “The United States of America National Report for the 2012 Convention on Nuclear Safety Extraordinary Meeting,” Revision 4, issued July 2012. This recommendation was identified as an activity that should begin without unnecessary delay (i.e., Tier 1).

The addressees were requested to assess their current communications systems and the equipment that would be used during an emergency event assuming that a large scale natural event resulted in a loss of all alternating current power on site, to consider enhancements regarding the communications requirement in NRC regulations (10 CFR 50.47, “Emergency Plans,” and Appendix E to 10 CFR Part 50) and in NUREG-0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981, and to assume the event resulted in extensive damage to normal and emergency communications systems both on site and in the area surrounding the site and that cellular and other communications infrastructures were unavailable. Addressees also were asked to evaluate how communications equipment used during an emergency event would be powered assuming a prolonged SBO. They were asked to evaluate their emergency response organization staffing following the occurrence of a large scale natural event that altered the normal access routes to the site, thereby affecting the response time for the emergency response organization. Addressees also were asked to assess both their current staffing levels and the appropriate staff and positions to respond to a multi-unit event given a beyond-design-basis natural event and to determine if enhancements were needed. Finally, addressees were requested to evaluate the minimum staffing that would be on site at the time the event occurred and to assess the need for additional onsite staff as the event unfolded, since this could affect a licensee’s assessment capabilities. At single unit sites, addressees were required to provide information on staffing necessary to cope with an extended loss of all alternating current power if access to the site was impeded.

On October 31, 2012, licensees provided communication assessments in response to the RFI. Based on review of the initial submittals, the NRC identified several generic issues and held a public meeting to discuss them with all involved stakeholders on January 3, 2013. In February 2013, licensees supplemented their October 2012 submittals to address the generic issues. The NRC staff continues to review these submittals.

On April 30, 2013, licensees provided a staffing assessment to respond to the first phase of the RFI. The staff is currently reviewing these submittals and expects to issue the results of the staff’s review no later than December 2013. The last phase of the staffing assessments has a dependency on NRC Order EA-12-049, “Order Modifying Licenses with Regard to Requirements for Mitigating Strategies for Beyond-Design-Basis External Events,” issued March 12, 2012. The staff expects to receive this assessment of the staffing four months before the second refueling outage at each site in accordance with EA-12-049.

Additionally, the NRC staff identified lessons learned applicable to the NRC Incident Response Program not covered under the NTTF recommendations. One of these items was the challenges faced in communicating with States and regional stakeholders. The staff has entered this item into the NRC's Incident Response Corrective Action Program and is currently implementing policy and procedure changes and intends to address this item as part of this effort.

The NRC is focusing its efforts on the implementation of the Tier 2 NTTF Recommendation 9.3, which includes:

- adding guidance to the emergency plan that describes the ability to perform a multi-unit dose assessment (including releases from SFP) using the licensees' site-specific dose assessment software and approach
- conducting periodic training and exercises for multi-unit and prolonged SBO scenarios
- ensuring that emergency preparedness equipment and facilities are sufficient for dealing with multi-unit and prolonged SBO scenarios

The NRC staff determined that the mitigating strategies recommendation (*Recommendation 4.2*) addresses periodic training and exercises for multi-unit and prolonged SBO scenarios and ensures that emergency preparedness equipment and facilities are sufficient.

The NRC staff considered options regarding how licensees should perform a multi-unit dose assessment using the licensee's site-specific dose assessment software and approach, and the NRC determined that an industry-proposed approach to have full capability to perform these dose assessments can be completed by the end of 2014.

In addition to the Tier 3 emergency preparedness items discussed in Section 1.3.2 of this report, the NRC staff also identified recommendations for lessons learned from the Fukushima event that may warrant regulatory action, but were not specifically included with the NTTF recommendations. The recommendations that require further staff study to support regulatory action (i.e., Tier 3) include:

- evaluate the basis of the emergency planning zone
- evaluate whether potassium iodine should be prestaged beyond the current 10 mile zone

There are plans to further study the potential health effects from the released radioactivity from the Fukushima site. The United Nations Scientific Committee on the Effects of Atomic Radiation plans a 2-year assessment of Fukushima impacts; the Fukushima Health Survey, a major initiative to inform future, more detailed dose assessments, will consider the location of every resident from the time the event occurred. NRC staff will monitor these efforts, and consider their implication for emergency planning around nuclear power plants in the United States. The NRC is conducting a Level 3 PRA to gain a better understanding of the radiological effects from postulated accident sequences, including those at multi-unit sites.

The NRC staff determined that the existing basis for the emergency planning zone size remains valid (even for multi-unit events). However, the NRC staff will use insights gained from the current Level 3 PRA study and information from the United Nations Scientific Committee on the Effects of Atomic Radiation assessment to evaluate the effect of a multi-unit event on the size of the emergency planning zone.

Regarding the prestaging of potassium iodide beyond the current 10-mile zone, NRC will continue to study the health effects on populations surrounding nuclear power plants, and will consider the population health studies performed by the Japanese Government, when they become available to determine if any policy changes are necessary.

ARTICLE 17. SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented for

(i) evaluating all relevant site-related factors that are likely to affect the safety of a nuclear installation for its projected lifetime

(ii) evaluating the likely safety impact of a proposed nuclear installation on individuals, society, and the environment

(iii) re-evaluating, as necessary, all relevant factors referred to in subparagraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation

(iv) consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, upon request, providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation

This section explains the NRC's responsibilities for siting, which include site safety, environmental protection, and emergency preparedness. First, this section discusses the regulations applying to site safety and their implementation, emphasizing regulations applying to seismic, geological, hydrological, meteorological, and radiological assessments. Next, it explains environmental protection, reevaluation of site-related factors, and lessons learned from Fukushima. Article 16 of this report discusses emergency preparedness and international arrangements, which would apply to Contracting Parties in obligation iv, above.

17.1 Background

The NRC's siting responsibilities stem from the Atomic Energy Act, the Energy Reorganization Act, and the National Environmental Policy Act. These statutes confer broad regulatory powers on the Commission and authorize the NRC to issue regulations that it deems necessary to fulfill its responsibilities under the acts.

As discussed in Article 7 of this report, in 1989 the NRC revised the regulatory approach governing the licensing of new nuclear power plants. This approach provides for certified standard designs and combined licenses that resolve design issues before construction, and early site permits that resolve most siting and environmental issues years before construction. To date, the NRC has issued four early site permits and two combined licenses using the revised regulatory approach in 10 CFR Part 52, with additional early site permits and combined license applications currently under review.

The NRC's siting regulations are integral to protecting public health and safety and the environment. Siting away from densely populated centers has been, and will continue to be, an essential component of the NRC's defense-in-depth safety philosophy (see Article 18 of this report), which also includes multiple-barrier containment and redundant and diverse safety systems. The primary factors that determine public health and safety are reactor design and construction and operation of the facility. However, siting factors and criteria are important to ensure that radiological doses from normal operation and postulated accidents will be acceptably low, natural phenomena and manmade hazards will be properly accounted for in the design and operation of the plant, and the human environment will be protected during the construction and operation of the plant.

17.2 Safety Elements of Siting

This section explains the safety elements of siting. After providing a short background, it explains the basic framework for assessing nonseismic, seismic, and other geological factors important to siting. Finally, it discusses radiological assessments performed for initial licensing, as a result of facility changes, and according to regulatory developments since the licensing of all U.S. operating plants.

17.2.1 Background

The NRC's site safety regulations consider societal and demographic factors, manmade hazards (such as airports and dams), and physical characteristics of the site (such as hydrological, seismological, and meteorological factors) that could affect the design or operation of the plant. Siting requirements for applications submitted after January 10, 1997, are specified in Subpart B, "Evaluation Factors for Stationary Power Reactor Site Applications on or after January 10, 1997," to 10 CFR Part 100, "Reactor Site Criteria." Siting factors that must be considered are specified in 10 CFR 100.20 and include population distributions, proximity to man-related hazards, and the physical characteristics of the proposed site. Nonseismic siting criteria in 10 CFR 100.21, "Non-Seismic Site Criteria," restrict occupancy around the site and establish limits on radiological releases and dose consequences from normal operations and postulated accidents. Geologic and seismic siting criteria in 10 CFR 100.23, "Geologic and Seismic Siting Criteria," require evaluation of all factors that might affect the design and operation of the proposed facility, and establish design bases for seismic and other naturally occurring phenomena.

To meet applicable regulatory requirements, the license applicant's safety analysis report must describe the physical characteristics in and around the site and contain accident analyses that are relevant to evaluating the suitability of a site. The NRC has developed numerous RGs to provide guidance on approaches that applicants can use to address issues of site safety and meet applicable requirements. The specifics of applicable RGs are discussed in subsequent sections of Article 17 of this report. RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 2, issued April 1998, provides a general set of safety and environmental criteria that the NRC staff has found useful in assessing candidate site identification in specific licensing cases. NUREG-0800 guides the staff in reviewing the site safety content of the applicant's safety analysis report. RS-002, "Processing Applications for Early Site Permits," dated May 3, 2004, identifies parts of NUREG-0800 that apply to the review of early site permits.

17.2.2 Assessments of Nonseismic Aspects of Siting

Siting facilities away from densely populated areas is a principal component of NRC's defense-in-depth safety philosophy. The evaluation of population distributions and the creation of restricted-use zones around a proposed facility are essential elements of compliance with regulatory requirements in 10 CFR Part 100. The dimensions of an inner "exclusion zone" and an outer "low population zone" will depend on plant design aspects such as the reactor power level and allowable containment leak rate, as well as the atmospheric dispersion characteristics of the site. In addition, the distance to a population center of more than about 25,000 residents must be at least 1.3 times the distance from the reactor to the outer boundary of the "low population zone." Radiological doses for postulated accidents are calculated using methods presented in Section 17.2.4 of this report. These doses are used to evaluate the effectiveness of the proposed restricted-use zones.

Accidents at nearby civilian or military facilities, or from nearby transportation routes, might produce missiles, shock waves, flammable vapor clouds, toxic chemicals, or incendiary fragments. These phenomena might affect the nuclear power plant itself or the plant operators in a way that jeopardizes the safety of the facility. As established in 10 CFR 100.21(e), potential hazards associated with these manmade features must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk to the proposed nuclear power plant. Additional information on the evaluation of these hazards is given in RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1, issued December 2001; RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1, issued February 1978; and RG 1.217, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," Revision 0, issued August 2011.

Radiological dose calculations must use meteorological data from the site. The site's atmospheric characteristics, combined with engineered safety features, must keep potential radiological doses from postulated accidents below the regulatory limits established in 10 CFR 50.34, "Contents of Applications; Technical Information." Acceptable approaches for obtaining meteorological data are given in RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 3, issued March 2007. These meteorological data also are used in safety analyses or to establish plant design bases for phenomena such as wind loads or impacts from tornado-generated missiles. RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007, and RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, issued October 2011, provide additional information on assessing these phenomena.

In siting a nuclear power plant, a highly dependable system of water supply sources should be available under postulated occurrences of natural phenomena and site-related accident phenomena. Considerations for water supply are addressed in RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, issued January 1976. Because of the likely proximity to water, many sites need to be evaluated for flood hazards from precipitation, wind, tsunami, or human-related hazards such as dam failure events. Acceptable approaches for conducting flood-hazard evaluations are given in RG 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 2, issued August 1977.

Site characteristics also are an important component of emergency and security planning. For emergency planning, 10 CFR 100.21 requires the site evaluation to determine whether there are any characteristics that would pose a significant impediment to taking protective actions to protect the public in the event of emergency. In addition, 10 CFR 100.21 also requires that site characteristics must allow for the development of adequate security plans and measures.

17.2.3 Assessments of Seismic and Geological Aspects of Siting

The NRC's siting regulations listed in Section 17.2.1 of this report detail the assessments applying to seismic and geologic aspects of siting. In simple terms, all geologic factors that might affect the design or operation of the nuclear power plant must be assessed. Recent developments in these geologic assessments include a performance-based approach for determining the site-specific ground motion response spectrum and the safe-shutdown earthquake. The performance-based approach described in RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," issued March 2007, combines the site seismic hazard curves and seismic fragility curves for nuclear structures to meet a specified performance target. RG 1.208 also incorporates recent developments in seismic hazard assessment, including the use of cumulative absolute velocity filtering in place of a lower-bound magnitude cutoff and guidance on the development of earthquake time histories, site response analysis, and the location of the ground motion response spectrum within the soil profile.

In 2012, a new seismic source model was completed for the central and eastern United States (NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," issued January 2012), which built upon previous seismic source models. The new seismic source model used a Senior Seismic Hazard Analysis Committee Level 3 assessment process to represent the center, body, and range of technically defensible interpretations of the available data, models, and methods (NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," issued April 1997). The updated model provides a consistent and stable basis for evaluating seismic source zones in probabilistic seismic hazards assessments for the central and eastern United States.

The NRC reviews and certifies new and advanced reactor designs under 10 CFR Part 52. The seismic capacity of the certified designs is determined independent of any specific site but capable of being sited in most currently existing sites. The NRC requires all new and advanced reactor designs to demonstrate that they have a plant-level seismic margin of 1.67 times the design-basis safe-shutdown earthquake with high confidence (i.e., 95 percent) in low (i.e., 5 percent) probability of failure. The design is confirmed to be suitable for the seismic hazard of the proposed construction site as part of the combined license review.

17.2.4 Assessments of Radiological Consequences from Postulated Accidents

The Reactor Site Criteria Rule, 10 CFR Part 100, contains provisions for assessing whether radiological doses from postulated accidents will be acceptably low. The NRC has issued the following regulatory guidance for licensees to implement the requirements for dose assessments from postulated accidents:

- RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors," Revision 2, issued June 1974

- RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors," Revision 2, issued June 1974

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued November 1982

In addition to regulatory guides, the NRC staff review guidance in NUREG-0800, Chapter 15, "Transient and Accident Analysis," provides additional information on analysis methods acceptable to the staff.

Regulations also require that, in addition to the analysis of internally initiated accident sequences, the potential hazards associated with nearby transportation routes and industrial and military facilities must be evaluated. Site parameters must be established so that potential hazards from such routes and facilities will pose no undue risk to the proposed nuclear power plant.

NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” issued February 1995, provides updated information on light-water reactor accident source terms. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” issued July 2000, provides guidance on the implementation of NUREG-1465. RG 1.183 presents one method that may be used to show compliance with 10 CFR 50.67, “Accident Source Term,” or the accident dose assessment requirements in 10 CFR 50.34 and 10 CFR Part 52 for new light-water reactor licensing.

Although applicants analyze dose primarily to support reactor siting, licensees are required to evaluate the potential increase in the consequences of accidents that might result from modifying facility SSCs. Commitments (including the radiological acceptance criteria) the applicant made during siting and documented in its final safety analysis report remain binding until modified. A licensee must evaluate the potential consequences of design changes against these radiological criteria to demonstrate that the changes will result in a design that still conforms to the regulations and commitments. If the consequences increase more than minimally, as outlined in 10 CFR 50.59, or require a change to the technical specifications, as discussed in Article 14 of this report, the licensee must obtain NRC approval before implementing the proposed modification. Requirements in 10 CFR 50.67 allow licensees to use an alternative source term in place of the accident source term used in the original licensing and siting of the operating facility.

If a licensee has not implemented the alternative source-term approach in 10 CFR 50.67, RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” issued May 2003, provides an acceptable approach for assessing the potential significance of changes to plant design and licensing bases. Thus, RG 1.195 provides an alternative approach to the dose assessment methods in RG 1.3 and RG 1.4.

The NRC has applied the 1996 revision to 10 CFR Part 100, along with the alternative source term as described in RG 1.183, in its design certification review for a passive advanced light-water reactor, the AP600. More recently, the agency has applied the practice to the AP1000 reactor with similar results and is applying it for all contemplated light-water reactor design certification application reviews, including the ESBWR, the U.S. EPR, and the U.S. APWR. For other than light-water reactor designs and advanced reactors, applicants will have to describe their rationale for an appropriate accident source term characterization that will be subject to NRC independent review.

The industry continues to explore the use of the alternative source term in implementing cost-beneficial licensing actions at operating reactors. Some of these applications resulted in improved safety equipment reliability and reduced occupational exposures. Since the issuance of 10 CFR 50.67, more than half of the operating reactor licensees requested either full implementation of the alternative source term or selective implementation for certain regulatory applications. Operating plant licensees also have used the alternative source term to analyze the adequacy of certain engineered safety features in meeting the operability requirements in their operating reactor technical specifications.

17.3 <u>Environmental Protection Elements of Siting</u>

This section explains the environmental protection elements of siting. It covers the governing documents and site approval process. Since the last operating plants in the United States received licenses, issues have arisen that must be considered in siting reviews, including reviews for new facilities proposed since 2003. This section explains the effect of these issues.

17.3.1 Governing Documents and Process

The environmental protection elements of siting consist of the plant's demands on the environment (e.g., water use and effects of construction and operation). These elements are addressed in 10 CFR Part 51, which implements the National Environmental Policy Act consistent with the NRC's statutory authority and reflects the agency's policy to voluntarily apply the regulations of the President's Council on Environmental Quality, subject to certain conditions. Integrating environmental reviews into its routine decisionmaking, the NRC considers environmental protection issues and alternatives before taking any action that may significantly affect the human environment.

The site approval process leading to the construction or operation of a nuclear power plant requires the NRC to prepare an environmental impact statement. The updated and revised environmental standard review plans (NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," issued March 2000) guide the staff's environmental reviews for a range of applications, including green field site reviews for construction permits and operating licenses under 10 CFR Part 50, for early site permits under 10 CFR Part 52, Subpart A, "Early Site Permits," and for combined licenses under 10 CFR Part 52, Subpart C, "Combined Licenses," when the application does not reference an early site permit. The NRC issued updates to review practices in 2007 and 2010 to reflect experience gained from early site permit reviews, account for the changes resulting from the amendment to the limited work authorization rule (discussed later in this section), and include consideration of the environmental effects of greenhouse gas emissions and climate change. Article 19 of this report discusses these governing documents and processes for combined license reviews.

Environmental standard review plans are also appropriate for environmental reviews of applications for combined licenses under 10 CFR Part 52, Subpart C, when the applications reference an early site permit. Reviews of early site permit applications are limited because the reviews focus on the environmental effects of reactor construction and operation that have characteristics that fall within the postulated site parameters and because the reviews need not assess benefits (e.g., the need for power) or alternative energy sources. The environmental information in applications for combined licenses that reference an early site permit is limited to (1) information to demonstrate that the design of the facility falls within the parameters specified in the early site permit, (2) new and significant information on issues previously considered in the early site permit proceeding, and (3) any significant environmental issue not considered in any previous proceeding on the site or design.

The environmental standard review plans in Supplement 1 to NUREG-1555 guide the staff's environmental review for license renewal applications under 10 CFR Part 54. Article 14 of this report discusses the license renewal process in more detail.

Several other NRC actions on siting and site suitability require environmental reviews, including issuance of limited work authorizations (10 CFR 50.10(e); 10 CFR 52.25, "Extent of Activities Permitted"; and 10 CFR 52.91, "Authorization to Conduct Site Activities"), early partial decisions (10 CFR 2.600, "Scope of Subpart," in Subpart F, "Additional Procedures Applicable to Early Partial Decisions on Site Suitability Issues in Connection with an Application for a Permit to Construct Certain Utilization Facilities," of 10 CFR Part 2, "Agency Rules of Practice and

Procedure”), and preapplication early reviews of site suitability issues (Appendix Q, “Pre-application Early Review of Site Suitability Issues,” to 10 CFR Part 50).

With its 2007 amendment to the limited work authorization licensing framework (10 CFR 50.10, “License Required; Limited Work Authorization”), the Commission limited its authority to construction activities that have a “reasonable nexus to radiological health and safety or common defense and security” and defined “construction” within the context of its authority. The effect of this change is not restricted to limited work authorizations. Other activities related to building the plant that do not require NRC approval (but may require a permit from other regulatory agencies) may occur before, during, or after NRC-authorized construction activities. These activities, called “preconstruction” in 10 CFR 51.45(c), may be regulated by other local, State, Tribal or Federal agencies. On September 12, 2008, the NRC and the U.S. Army Corps of Engineers signed an updated memorandum of understanding to enhance the effectiveness of reviews of nuclear power plant license applications that would require multiple Federal permits under separate statutes. The NRC and the U.S. Army Corps of Engineers are participating as cooperating agencies in the preparation of many environmental impact statements.

17.3.2 Other Considerations for Siting Reviews

Since the NRC last issued construction permits under 10 CFR Part 50 in the 1970s and coincident with the publication of the initial environmental standard review plan, many changes to the regulatory environment have affected the NRC and applicants seeking site approvals. These include new environmental laws and regulations, changes in policies and procedures resulting from decisions of courts and administrative hearing boards, and changes in the types of authorizations, permits, and licenses issued by the NRC. This section highlights some of these changes and their effects on the environmental standard review plans.

In the late 1980s, the NRC issued regulations that gave an alternative licensing framework to 10 CFR Part 50, which required a construction permit followed by an operating license. The new framework in 10 CFR Part 52 introduced the concepts of approving designs independent of sites and approving sites independent of designs, and then efficiently linking these approvals to approve construction and operation of the facility. As discussed in the introduction of this report, the NRC completed several reviews of early site permits and combined license applications under 10 CFR Part 52 and is actively conducting additional siting and new plant licensing reviews.

As part of the revisions to the licensing framework, the NRC issued RS-002, which embodies the environmental guidance in NUREG-1555, the environmental standard review plan, and the outcome of interactions with stakeholders. In addition, in 2007, the NRC revised 10 CFR Part 52 to reflect experience gained in its use and to provide guidance on the preparation of combined license applications. As part of that rulemaking the NRC issued RG 1.206, “Combined License Applications for Nuclear Power Plants,” in June 2007, which includes guidance on the assessment of environmental issues.

Since 1984, the NRC has considered the environmental impacts of spent nuclear fuel storage after the licensed lifetime of reactor operations to be a generic issue that is best addressed through rulemaking. Several technical concerns were identified in the analyses supporting the regulation that addressed this issue (10 CFR 51.23, “Temporary Storage of Spent Fuel after Cessation of Reactor Operation—Generic Determination of No Significant Environmental Impact”), which resulted in the U.S. Court of Appeals vacating this regulation in June 2012. The NRC is developing an environmental impact statement to address the technical concerns raised by the Court and provide the National Environmental Policy Act analyses needed to support a revision to 10 CFR 51.23. Although licensing reviews and hearings are continuing while these rulemaking activities are ongoing, the NRC will not issue licenses dependent on 10 CFR 51.23 before these concerns with long-term spent fuel storage are resolved.

As described in previous U.S. National Reports, other relevant regulatory developments include the following:

- Presidential Executive Order 12898, "Federal Actions To Address Environmental Justice in Minority and Low-Income Populations," issued February 1994, which instructed Federal agencies to make "environmental justice" part of each agency's mission by addressing disproportionately high and adverse human health or environmental effects of Federal programs, policies, and activities on minority and low-income populations

- the Yellow Creek Decision, which determined that the authority of the NRC is limited in matters that are expressly assigned to EPA

- changes in the economic regulation of utilities that have expanded the options to be addressed in considering the need for power in environmental impact statements

- design alternatives to mitigate the consequences of severe accidents

- EPA rules about cooling water intake structures

17.4 Re-evaluation of Site-Related Factors

Although operating nuclear power plants are not reevaluated periodically for site-related factors, the continued safety of nuclear plants and the adequate protection of a licensed plant are imperative. If there is a significant change in any hazard to an already licensed nuclear plant, then the NRC will determine whether a backfit action under 10 CFR 50.109 is necessary. The NRC will always require the backfitting of a nuclear power plant if it determines that such regulatory action is necessary to ensure that the plant provides adequate protection to the health and safety of the public and is in accordance with the common defense and security.

In response to the Fukushima accident, the NRC used its regulatory processes to request that licensees reevaluate the seismic and flooding hazards at their sites using present-day regulatory guidance and methodologies and, if necessary, to perform a risk evaluation. The results of these reevaluations will be used to determine whether additional regulatory actions are necessary to ensure plants are adequately protected from seismic and flooding events.

Periodic seismic requalification of equipment is not necessary, because databases are available for equipment already qualified or tested to fragility levels. IEEE standard 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," provides criteria to determine the appropriate level of equipment ruggedness. Using this standard, a licensee is able to determine whether equipment needs to be requalified or replaced.

17.5 Consultation with Other Contracting Parties To Be Affected by the Installation

At this time, the NRC does not have any specific international arrangements with neighboring States for siting new builds. However, the agency's current arrangements with its Canadian and Mexican regulatory counterparts for the exchange of information and experience would serve as the mechanism for any cooperative dialogue if such a situation arose.

17.6 Fukushima Lessons Learned

As discussed earlier in this Article, the NRC's current regulatory framework for facility siting is protective of public health and safety, and has sufficient flexibility to accommodate new information that may result from continued examination of the nuclear accident at Fukushima, as well as the reevaluations of seismic and flooding hazards already underway at each site in the U.S. Fukushima lessons learned siting considerations in plant design are discussed in Article 18.

ARTICLE 18. DESIGN AND CONSTRUCTION

Each Contracting Party shall take the appropriate steps to ensure that:

(i) the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur

(ii) the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis

(iii) the design of a nuclear installation allows for reliable, stable, and easily manageable operation, with specific consideration of human factors and the man-machine interface

This section explains the defense-in-depth philosophy and how it is embodied in the general design criteria of U.S. regulations. It explains how applicants meet the defense-in-depth goals and how the NRC reviews applications and conducts inspections before issuing licenses to ensure that this philosophy is implemented in practice. Next, this section discusses measures for ensuring that the applications of technologies are proven by experience or qualified by testing or analysis. Finally, this section discusses requirements for reliable, stable, and easily manageable operation, specifically considering human factors and the man-machine interface, and lessons learned from Fukushima. Article 12 of this report also provided information on the human factors obligations.

18.1 <u>Defense-in-Depth Philosophy</u>

This section explains the defense-in-depth philosophy followed in regulatory practice, governing documents and regulatory process for designing and constructing a nuclear power plant. It also discusses relevant experience and examples.

18.1.1 Governing Documents and Process

The defense-in-depth philosophy, as applied in regulatory practice, requires that nuclear plants contain a series of independent, redundant, and diverse safety systems. The physical barriers for defense-in-depth in a light-water reactor are the fuel matrix, the fuel rod cladding, the primary coolant pressure boundary, and the containment. The levels of protection in defense-in-depth are (1) a conservative design, quality assurance, and safety culture, (2) control of abnormal operation and detection of failures, (3) safety and protection systems, (4) accident management, including containment protection, and (5) emergency preparedness.

Appendix A to 10 CFR Part 50 embodies the defense-in-depth philosophy. General design criteria cover protection by multiple fission product barriers, protection and reactivity control systems, fluid systems, containment design, and fuel and radioactivity control. The NRC staff amplified its defense-in-depth philosophy in RG 1.174, which provides guidance on using a PRA in risk-informed decisions on plant-specific changes. The general design criteria establish the minimum requirements for the principal design criteria, which in turn establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs that are important to safety.

To ensure that a plant is properly designed and built as designed, that proper materials are used in construction, that future design modifications are controlled, and that appropriate maintenance and operational practices are followed, a good quality assurance program is needed. To meet this need, General Design Criterion 1, “Quality Standards and Records,” of Appendix A to 10 CFR Part 50, and its implementing regulatory requirements specified in Appendix B to 10 CFR Part 50, establish quality assurance requirements for all activities affecting the safety-related functions of the SSCs.

In accordance with the two-step licensing process under 10 CFR Part 50, an applicant for a construction permit must present the principal design criteria for a proposed facility in its preliminary safety analysis report. For guidance in writing a safety analysis report, the applicant may use RG 1.70. The safety analysis report also must contain design information for the proposed reactor and comprehensive data on the proposed site. The report must also discuss various hypothetical accident situations and the safety features to prevent accidents or, if accidents occur, to mitigate their effects on both the public and the facility’s employees.

After obtaining a construction permit under 10 CFR Part 50, the applicant must submit a final safety analysis report to support an application for an operating license, unless it submitted the report with the original application. This report should give the details of the final design of the facility, plans for operation, and procedures for coping with emergencies. The preliminary and final safety analysis reports are the principal documents the applicant provides for the staff to determine whether the proposed plant can be built and operated without undue risk to the health and safety of the public. Current applications to build new nuclear power plants have been submitted using the combined license process under 10 CFR Part 52, although applicants are not precluded from using the two-step licensing process under 10 CFR Part 50. Applications submitted under 10 CFR Part 52 must meet all of the 10 CFR Part 50 requirements as well as the applicable requirements referenced in other regulations (e.g., 10 CFR Part 20, Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” Part 40, “Domestic Licensing of Source Material,” Part 70, Part 73, and Part 100). The NRC issued guidance for the content and format of a combined license application in RG 1.206. A significant difference in the 10 CFR Part 52 process is that the final safety analysis report must be submitted before authorization is granted to begin construction. Article 19 of this report describes the combined license review process.

The NRC staff reviews safety analysis reports according to NUREG-0800 to ensure that the applicant has satisfied the general design criteria and other applicable regulations. The staff reviews each application to determine whether the plant design meets the Commission’s regulations (10 CFR Parts 20, 50, 73, and 100). These reviews include, in part, the characteristics of the site. In addition, each application for a nuclear installation must include a comprehensive environmental report that provides a basis for evaluating the environmental impact of the proposed facility. RG 4.2, Revision 2, gives applicants information on writing environmental reports. The NRC staff reviews the environmental reports according to NUREG-1555. In reviewing an application, the staff, supported by outside experts, conducts independent technical studies to review certain safety and environmental matters. The staff states its conclusions in an environmental impact statement and a safety evaluation report, which it may update before granting the license. Under the two-step licensing process in 10 CFR Part 50, the NRC does not issue an operating license until construction is complete and the Commission makes the findings required under 10 CFR 50.57, “Issuance of Operating

License." For combined license applications submitted under 10 CFR Part 52, the Commission must make a finding in accordance with 10 CFR 52.97, "Issuance of Combined License," to issue the combined license. With issuance of the combined license, construction of the facility may begin; however, the Commission must make a finding in accordance with 10 CFR 52.103(g) that all acceptance criteria in the combined license are met to authorize operation of the facility.

The NRC monitors nuclear power plant construction to ensure compliance with the agency's regulations to protect public health and safety and the environment. The NRC has developed an inspection program for nuclear plants licensed under 10 CFR Part 52.

The new inspection program revises the 10 CFR Part 50 Construction Inspection Program. It incorporates inspections, tests, analyses, and acceptance criteria (ITAAC) from 10 CFR Part 52, as well as lessons learned from the inspection program used in the previous construction era (1970-1980). It also considers modular construction at remote locations.

Before the combined license is issued, the NRC inspection program focuses on the applicant's establishment of a quality assurance program to verify that applications submitted to the NRC meet specified requirements in 10 CFR Part 52.

In addition, the inspection program focuses on supporting the NRC staff's preparation for the mandatory Atomic Safety and Licensing Board hearing and the final Commission decision on whether a combined license should be granted. Inspection Manual Chapter 2502, "Construction Inspection Program: Pre-Combined License (Pre-COL) Phase," dated December 13, 2010, lists inspections for this phase.

The NRC also interacts with manufacturers and suppliers of safety-related components through the NRC vendor inspection programs that inspect compliance with quality assurance and defect reporting requirements. Vendor inspections are conducted at vendor shops principally to examine whether the vendor has been complying with Appendix B to 10 CFR Part 50, as required by procurement contracts with applicants and licensees. Inspection Manual Chapter 2507, "Construction Inspection Program: Vendor Inspections," dated April 25, 2011, lists inspections for vendors.

During construction, NRC inspectors sample the spectrum of the applicant's activities related to the ITAAC in the combined license to confirm that the applicant is adhering to quality and program requirements. The NRC staff will verify successful ITAAC completion based on these inspections and will review all ITAAC closure notifications from the licensee. The NRC will publish notices in the *Federal Register* of completed ITAAC. Additionally, regional specialists inspect and monitor activities at the construction sites. The NRC will increase the number of resident inspectors stationed in construction sites. The NRC expects that the peak resident staffing will be approximately five inspectors at construction sites with one unit and seven at construction sites with two units. Inspection Manual Chapter 2503, "Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work," dated July 5, 2012, lists inspections for this phase.

In addition to inspections of ITAAC related work, the NRC inspection program addresses inspections of programs that support construction activities (e.g., quality assurance and preoperational testing) as well as programs that support eventual operation of the facility (e.g., fire protection, security, training, radiation protection, and startup testing), and programs that enable the transition of the organization from construction to power operations. Inspection

Manual Chapter 2504, “Construction Inspection Program—Inspection of Construction and Operational Programs,” dated October 24, 2012, lists inspections for this phase.

18.1.2 Experience

The agency’s ongoing efforts in reviewing the Watts Bar Nuclear Plant, Unit 2, license application is an example of how the NRC design and construction process for a 10 CFR Part 50 application (described in Section 18.1.1) is currently implemented.

The Watts Bar Nuclear Plant, owned by TVA, is located in southeastern Tennessee. The site has two Westinghouse designed PWRs. Watts Bar Unit 1 received a full-power operating license in early 1996, and it was the last new power reactor licensed in the United States under 10 CFR Part 50. Although TVA stopped construction at Watts Bar Unit 2 in the mid-1980s, it has now resumed Watts Bar Unit 2 construction, and its operating license application is currently pending before the Atomic Safety and Licensing Board. The construction permit for Watts Bar Unit 2 is currently active and expires in 2013. Because of delays in the project schedule, TVA has requested to extend the construction permit expiration date to 2016, and the NRC is currently reviewing this request.

In its regulatory framework for the completion of Unit 2, the Commission approved (SRM-SECY-07-0096, “Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant ,Unit 2,” dated July 25, 2007) a licensing review approach that uses the current licensing basis for Watts Bar Unit 1 as the reference basis for review and licensing of Unit 2. This approach will ensure safety while preserving design and operational consistency between the units. However, considering the construction status of the unit, the NRC encouraged TVA to adopt updated standards wherever feasible and look for opportunities to resolve any generic safety issues in which the unirradiated state of Unit 2 makes the issue easier to resolve before plant operation. The NRC’s operating license review will include safety design, environmental review, and inspection of construction activities.

TVA has updated its initial 1970s operating license application. The NRC has published a notice of the updated application for an operating license in the *Federal Register* to provide public notice and an additional opportunity for a hearing. Southern Alliance for Clean Energy had an admitted contention for hearing before an Atomic Safety and Licensing Board. In July 2013, this contention was dropped by Southern Alliance for Clean Energy. To date, Southern Alliance still has a contention held in abeyance regarding the Waste Confidence Rule. TVA has submitted its final supplemental environmental impact statement for the completion and operation of Watts Bar Unit 2. The staff published its draft supplemental environmental statement for completion and operation of Watts Bar Unit 2 in late 2011 for public comment. The final environmental statement was published in June 2013. The NRC also has held public outreach meetings in the vicinity of the site to inform the public about its licensing and inspection activities, including how the public can monitor and participate in the licensing process.

The NRC has established a dedicated team at both its headquarters and regional offices for review and inspection of the Unit 2 activities. The staff has independently reviewed TVA’s regulatory framework and documented its results in a safety evaluation report, NUREG-0847, Supplement 21, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2,” issued February 2009. The review identified the items that must be completed before issuance of an operating license. To date, the staff has published five additional supplements to NUREG-0847 documenting its review of the items laid out in Supplement 21. The NRC Region II office is performing necessary inspections and oversight activities. It developed

Inspection Manual Chapter 2517, “Watts Bar Unit 2 Construction Inspection Program,” issued February 2008, to provide guidance for these inspection activities. The NRC Region II office is examining historical inspection records, employee concerns, operating experience, scope of new or rework, and construction deficiency reports. The NRC has established a resident inspector office, with three resident inspectors dedicated to performing inspections at Watts Bar Unit 2.

As always, safety is the NRC’s main focus. Before issuing an operating license, the NRC will confirm that TVA has safely designed and constructed Watts Bar Unit 2 in accordance with regulatory requirements, and that the facility can be safely operated.

The NRC has established a Web site for its Watts Bar Unit 2 activities, which can be accessed at http://www.nrc.gov/info-finder/reactor/wb/watts-bar.html.

18.2 Technologies Proven by Experience or Qualified by Testing or Analysis

In 10 CFR 50.43(e), the NRC requires that new technologies are demonstrated to be proven. This rule requires demonstration of new technologies through analysis, appropriate test programs, experience, or a combination thereof. In its safety analysis reports for the AP600 and AP1000 standard plant designs, Westinghouse used separate effects tests, integral systems tests, and analyses to demonstrate that its passive safety systems will perform as predicted. Section 14.2 of this report discusses the qualification of currently used technologies.

18.3 Design for Reliable, Stable, and Easily Manageable Operation

The NRC specifically considers human factors and the human-system interface in the design of nuclear installations. For safety analysis reports, the NRC reviews the human factors engineering design of the main control room and the control centers outside of the main control room. Article 12 of this report also discusses human factors.

18.3.1 Governing Documents and Process

To support its reviews of the human factors engineering issues associated with the certification and licensing of new plant designs, the NRC uses NUREG-0800, Chapter 18, Revision 2, and NUREG-0700, “Human-System Interface Design Review Guidelines,” Revision 2, issued May 2002. The NRC used NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 2, issued February 2004, for evaluating the design of next-generation main control rooms listed in Section 18.1.2.2. In November 2012, the NRC issued NUREG-0711, Revision 3, “Human Factors Engineering Program Review Model,” to address lessons learned from these reviews. NUREG-0800, Section 14.3.9, “Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007, provides additional guidance. The NRC has recently initiated work to update these review guidelines. Additionally, the NRC developed guidance for reviewing combined license applications, RG 1.206, which includes sections that address the human factors engineering review of combined license applications.

18.3.2 Experience

The NRC is actively reviewing new plant designs and combined license applications.

18.3.2.1 *Human Factors Engineering*

The NRC has completed the evaluation of the human factors engineering sections of the design certification reviews of the ESBWR and AP-1000 applications as well as the Vogtle and V.C. Summer combined license applications. Reviews continue on the US APWR and US EPR design certification submittals and on the remaining combined license applications. The NRC's human factors engineering reviews for design certification applications principally focus on evaluating implementation plans for the design of the control facilities to ensure that the design process will be carried out consistent with state-of-the-art human factors principles. The NRC will verify acceptable implementation of these plans through specified ITAAC (i.e., design acceptance criteria).

The completed staff reviews identified the following weaknesses in the previous revision of NUREG-0711:

- The "human reliability analysis" element did not address manual actions credited in the Standard Review Plan, Chapters 7 and 15.
- The technical support facility, emergency operating facility, and local control stations are included in the human factors engineering program scope, but it was unclear which elements applied to them.
- The "verification and validation" element was complex and created confusion on how performance measurement criteria were meant to be applied.
- The content of Implementation Plans and Results Summary Reports were not adequately defined, resulting in insufficient detail in applications, confusion on which design products could be deferred, and difficulty in establishing ITAACs with sufficient scope.

NUREG-0711, Revision 3 was issued to address these issues.

18.3.2.2 Digital Instrumentation and Controls

RG 1.206 provides guidance for preparing the application for a combined license. Information in this RG is reflected in NUREG-0800. Chapter 7 of NUREG-0800 provides guidance to the NRC staff in reviewing the instrumentation and control design of the nuclear power reactors. This guidance assists the staff in determining whether the design complies with the applicable regulatory requirements and whether the applicant has demonstrated with reasonable assurance that the design provides adequate protection of public health and safety. All of the new reactor designs, such as the AP1000, ABWR, US EPR, ESBWR, and US APWR, contain highly integrated digital instrumentation and control systems, which present issues that are not relevant to analog systems. Examples of these issues include:

- A common-cause failure attributable to software errors was not possible with analog systems. This potential weakness may require consideration of diversity and defense-in-depth in the application of digital instrumentation and control systems.

- Digital system architectures raise issues such as interchannel communication, communication between nonsafety and safety systems, and cyber security that must be addressed to ensure that public safety is preserved.

- Highly integrated control room designs with safety and nonsafety displays and controls are the norm for new reactor designs. Human factors design and quality assurance during all phases of software development, control, and validation and verification are critical.

The NRC developed several ISG documents for review of new and innovative digital instrumentation and control systems found in new reactor designs. The guidance also provides the industry with the expectations and criteria the staff uses to evaluate their designs and determine compliance with NRC regulations. The staff is using this guidance in its review of applications for design certifications and combined licenses. The staff is in the process of incorporating the interim staff guidance into formal NRC staff guidance in NUREG-0800 and associated RGs. All ISGs on digital instrumentation and control can be found at http://www.nrc.gov/reading-rm/doc-collections/isg/digital-instrumentation-ctrl.html. DI&C-ISG-02, "Diversity and Defense-in-Depth," Revision 2, issued June 2009, has already been incorporated into NUREG-0800, Chapter 7, Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," Revision 6, issued July 2012.

The staff has completed its safety reviews of the instrumentation and control systems for the AP1000, ESBWR, and ABWR reactor designs, and is in the process of reviewing the instrumentation and control design for the US EPR and US APWR reactor designs. The staff also has initiated the instrumentation and control ITAAC inspection activities for the AP1000 combined licenses, and preapplication activities on the APR-1400 and small modular reactor designs.

To prepare for the review of applications for small modular reactor design certifications and combined licenses, the NRC staff is developing a design-specific review standard. Chapter 7 of the design-specific review standard is being developed as an innovative initiative specifically for the mPower™ design. This design-specific review standard chapter provides guidance to the staff for reviewing the instrumentation and control design of the Babcock & Wilcox mPower™ nuclear power reactor. This guidance will assist the staff in determining whether the design complies with the applicable regulatory requirements and whether the applicant has demonstrated with reasonable assurance that the design will provide adequate protection of public health and safety. This design-specific review standard chapter reflects a number of important lessons the staff learned when using NUREG-0800 to review new large light water reactor designs. The staff has incorporated the following lessons learned into this guidance to:

- Emphasize fundamental instrumentation and control design principles such as independence, redundancy, determinism, and diversity and defense-in-depth, as derived through design and analysis, such as hazard analysis, to prevent loss or impairment of a safety function. This guidance aims to address all of the significant aspects of the instrumentation and control design in a unified manner through this framework.

- Reflect an integrated instrumentation and control design using digital technology, which is common in new and advanced reactor designs. In addition, the topical areas most significant to safety are discussed first. The NUREG-0800 guidance is system-based; therefore, many regulatory requirements and their supporting guidance are repeated in multiple subsections. The approach of this design-specific review standard minimizes such repetition.

- Introduce the use of an integrated hazards analysis approach, which is a well-established safety engineering practice. This approach consolidates the various methods discussed in the current NUREG-0800 and provides a consistent, comprehensive, and systematic way to address the potential hazards associated with instrumentation and control systems in a unified framework.

- Address various new sources, such as the Multinational Design Evaluation Program common positions and lessons learned from other countries.

- Encompass all relevant branch technical positions contained in the current NUREG-0800. This guidance also clarifies the interface between the instrumentation and control area and other disciplines, such as human factors engineering, quality, and reactor systems.

The NRC participates in the Multinational Design Evaluation Program, an international assembly of nuclear regulators addressing common issues with the licensing of new reactors. The NRC chairs the Digital Instrumentation and Control Issue-Specific Working Group, which is looking at ways to harmonize requirements, standards, and guidance for instrumentation and control. The NRC is also working with the EPR digital instrumentation and control technical expert subgroup, which is an international collaboration of regulatory agencies engaged in review of the EPR instrumentation and control design. The Multinational Design Evaluation Program allows the NRC to share digital instrumentation and control information to support regulatory infrastructure improvements and licensing decisions.

18.3.2.3 Cyber Security

After September 11, 2001, the NRC issued two security-related orders, NRC Order EA-02-026, "Issuance of Order for Interim Safeguards and Security Compensatory Measures," issued February 2002, and NRC Order EA-03-086, "Issuance of Order Requiring Compliance with Revised Design Basis Threat for Operating Power Reactors," issued April 2003, which require power reactor licensees to implement measures to enhance cyber security. These security measures required immediate identification and assessment of computer-based systems deemed to be critical to the operation and security of the facility. From 2006 through February 2009, cyber security design reviews were performed solely based on the guidance in RG 1.152, Revision 2.

Subsequently, in March 2009, the NRC issued a new rule on cyber security, 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks," and RG 1.152 was revised to remove cyber security guidance. The cyber security rule requires licensees to provide high assurance that nuclear power plants' safety-related, important-to-safety, security, and emergency preparedness functions are protected from cyber attacks up to and including the design-basis threat. This new regulation required licensees and combined license applicants to submit a cyber security plan, including an implementation schedule, to the NRC for review and approval. Operating reactor licensees were required to submit a cyber security plan by November 23, 2009, and combined license applicants are required to submit a plan in accordance with their overall license application. Essential elements of a plan include describing the process for finding critical digital assets, describing the defensive model (i.e., protective strategy), referencing a comprehensive set of security controls, and describing the process for addressing each control. The cyber security plan also must acknowledge a commitment to maintain the cyber security program and provide adequate documentation of how that will be accomplished.

In January 2010, the NRC published RG 5.71, "Cyber-Security Programs for Nuclear Facilities," which provides implementation guidance to licensees and applicants on an acceptable method for satisfying the requirements of 10 CFR 73.54. This guidance describes an acceptable method licensees can follow to address potential security vulnerabilities in each life-cycle phase of critical digital assets that perform safety-related, important-to-safety, security, and emergency preparedness functions. It is equally applicable to both combined license applicants and the current fleet of operational reactors. The guidance embodies recommended practices from standards organizations such as the International Society of Automation, IEEE, the National Institute of Standards and Technology, and DHS.

In January 2010, the NRC and the North American Electric Reliability Corporation entered into a 5–year memorandum of understanding to address nuclear plant cyber security roles, responsibilities, and areas of coordination between the two organizations. Subsequent to the memorandum of understanding with the North American Electric Reliability Corporation, the NRC determined that 10 CFR 73.54 should be interpreted to include SSCs that have a nexus to radiological health and safety at NRC-licensed nuclear power plants. The Federal Energy Regulatory Commission and the North American Electric Reliability Corporation found this policy decision acceptable and they, likewise, found the NRC's regulatory framework sufficient to meet the North American Electric Reliability Corporation cyber security requirements for power generation plants. In accordance with the memorandum of understanding, the staff will continue to coordinate with the North American Electric Reliability Corporation to The memorandum of understanding recognizes the need for coordination, information sharing, and incident management and response between the two organizations. share relevant operating experience and other related technical information. The NRC also has a Memorandum of Agreement with the Federal Energy Regulatory Commission to facilitate a continuing and cooperative relationship and the exchange of experience, information, and data related to the reliability of the U.S. bulk electricity supply.

Presently, licensees are working to implement their cyber security programs, and the NRC has developed inspection guidance documents to verify compliance with the approved cyber security plans. Work to improve these guidance documents continues. Oversight activities that the NRC performs, including cyber security inspections, are being conducted by trained and qualified headquarters and regional NRC inspectors.

The NRC has also implemented a significant and continuing research program in cyber security for digital plant control systems.

18.4 New Reactor Construction Experience Program

The nuclear industry in the United States faced many construction quality and design issues in the 1970s and 1980s. In 1984, the NRC issued NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants," to document the lessons learned from plant construction. Since then, the NRC has revised some of its licensing review processes and construction oversight programs to implement recommendations made in NUREG-1055. In 2007, the NRC began developing a construction experience (ConE) program to focus on collecting, analyzing, and applying lessons learned from the design and construction of new reactors. To achieve this goal, the NRC staff developed a risk-informed process to obtain, screen, evaluate, communicate, and incorporate construction experience insights into its new reactor licensing and construction oversight activities.

Since 2007, the NRC staff has actively obtained and evaluated ConE information from various domestic and international sources. The ConE program also reviews all of the operating experience from operating reactors, because the root causes of many events at currently operating reactors date back to their design and construction period. To make the ConE information available and accessible to all NRC staff members, including technical reviewers located at NRC headquarters and inspectors located in regional offices, the staff is integrating

ConE information into the existing agency OpE database. This database enables all NRC staff to search and retrieve ConE information through word search, plant information, technical discipline, and other methods. NRC staff review of many of the construction-related issues has resulted in numerous generic communications in the form of INs to communicate lessons learned to the public and industry. A specific NRC task force evaluated the Fukushima event and lessons learned were acted on separate from the ConE program.

Many of the significant construction and operating events and issues discussed throughout this report have been evaluated by the ConE program for lessons learned. This includes the Seabrook alkali silica reaction phenomenon, the North Anna earthquake, flooding at Fort Calhoun, and design control during construction. The staff routinely and periodically updates the NEA ConEx database with many of these events, including NRC ConE lessons learned.

The NRC staff values close cooperation with the international community for the exchange of information on design and construction of new reactors. The NRC program has been working closely with several countries that are currently building new nuclear power plants. These interactions are carried out through established agency bilateral and multilateral agreements with other countries. For example, the NRC ConE program staff is contributing to the work of the NEA working group on regulation of new reactors, working group on operating experience, and the European Commission Joint Research Center. The NRC ConE staff also visits international sites under construction every year to further its cooperation and exchange of technical and regulatory information with other regulatory agencies. For instance, China's National Nuclear Safety Administration, the French Nuclear Safety Authority, and the Finnish Radiation and Nuclear Safety Authority have hosted a number of NRC inspectors at their new reactor construction sites over the last several years. These interactions have provided an exceptional hands-on experience for the NRC inspectors to gain a better understanding of the regulatory process and the construction inspection activities in these countries. Similarly, the NRC has hosted several staff members from foreign nuclear safety regulatory agencies, such as those from China, Republic of Korea, the United Arab Emirates, and France, to provide an opportunity for the agency's international counterparts to observe and learn about the licensing process and the oversight of new reactor construction activities in the United States. The NRC values such partnerships with other regulatory agencies and is committed to continuing its collaborative relationship with the international community to promote nuclear safety, security, and protecting people and the environment.

18.5 Fukushima Lessons Learned

The NRC has long recognized that protection from natural phenomena is an important means to prevent core damage and ensure the integrity of containment and the SFP. The NRC established several requirements addressing natural phenomena in 1971 with General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50.

General Design Criterion 2 requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena, such as floods, tsunamis, and seiches, without losing the capability to perform their safety functions. General Design Criterion 2 also requires that design bases for these SSCs reflect (1) appropriate consideration of the most severe of the natural phenomena that historically have been reported for the site and surrounding region, with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which the data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Since the establishment of General Design Criterion 2, the NRC's requirements and guidance for protection from seismic events, floods, and other natural phenomena have continued to evolve. The NRC has developed new regulations, new and updated regulatory guidance, and several regulatory programs to enhance previously licensed reactors, including the following:

- Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100
- NUREG-0800 and ISG in three areas related to protection from natural phenomena
- The systematic evaluation program established in 1977 to review the designs of older operating nuclear reactor plants to reconfirm and document their safety
- GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46," dated February 19, 1987, addressing concerns related to seismic qualification of mechanical and electrical equipment in operating nuclear power plants
- Supplement 4 to GL 88-20, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities, 10 CFR 50.54(f) (Subsection (f) of 10 CFR 50.54, "Conditions of Licenses')," dated June 28, 1991, requesting licensees to perform an individual plant examination of external events to identify vulnerabilities
- Subpart B, "Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997," to 10 CFR Part 100
- Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50

The staff also has published several RGs that address specific technical issues related to protection from natural phenomena, including RG 1.208, RG 1.221, and the following:

- RG 1.29, "Seismic Design Classification," issued March 2007
- RG 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 2, issued August 1977
- RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," issued December 1973
- RG 1.102, "Flood Protection for Nuclear Power Plants," Revision 1, issued September 1976
- RG 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants," issued March 2009

The NRC staff continually evaluates new information on natural phenomena, including lessons learned from operational experience and their potential impact on risk and overall plant safety. These evaluations have led to new requirements or guidance as discussed above, updated regulatory guidance, generic communications, and plant-specific actions to address identified issues.

At the time of the Fukushima accident, the NRC staff was proceeding with regulatory actions to request licensees to evaluate updated seismic hazard information. In support of early site permits and combined license applications for new reactors, the NRC staff reviewed updates to the seismic source and ground motion models provided by the applicants. The NRC reviews of the applications identified higher seismic hazard estimates than previously assumed, increasing the likelihood of exceeding the safe-shutdown earthquake at operating facilities in the Central and Eastern United States. In 2005, the staff recommended an examination of increased seismic hazard estimates in the central and eastern United States under the NRC Generic Issues Program. The NRC established Generic Issue (GI) 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," on June 9, 2005.

In 2010, the NRC concluded that GI-199 should transition to the regulatory assessment stage of the Generic Issues Program. IN 2010-018, "Generic Issue 199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated September 2, 2010, summarizes the results of the GI-199 safety and risk assessment. After it issued the IN, the NRC asked licensees to evaluate the updated seismic hazard analysis. The staff recommended, and the Commission approved, the incorporation of GI-199 into the regulatory actions being taken within the context of the Japan Lessons-Learned Project Directorate.

As a result of the NTTF review of the Fukushima events, the NRC concluded that seismic and flooding hazards warranted further consideration because of significant advancements in the state of knowledge and state of analysis in these areas in the time period since the operating plants were sited and licensed. One example of advancement in the state of knowledge is NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," issued January 2012, which presents updated composite seismic hazard curves for the central and eastern United States that resulted from a joint NRC, DOE and EPRI effort. The Japan earthquake and subsequent tsunami also highlighted the need to evaluate concurrent related events, such as seismically induced fires and floods. The NTTF identified a number of regulatory actions in the area of external events. These actions, as expanded or modified by the Japan Steering Committee, are discussed below.

18.5.1 Seismic, Flooding, and Other Hazards Protection

The NRC is undertaking near-term regulatory activities to reevaluate and upgrade, as necessary, the protection of SSCs against design-basis seismic and flooding events for all operating reactors in the United States. These activities are based on NTTF Recommendations 2.1 and 2.3, as modified by subsequent NRC management direction. These activities include (1) requesting that licensees reevaluate the seismic and flooding hazards, and (2) perform inspections (or "walkdowns") of important safety-related systems and components to identify plant-specific vulnerabilities.

The NRC, through TI 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event," dated March 23, 2011, performed plant inspections that provided information about the readiness of licensees to respond to seismic and flooding events. In response to Bulletin 2011-01, "Mitigating Strategies," dated May 11, 2011, the licensees verified that mitigating strategy equipment is capable of performing the required actions. The licensees also reported on the maintenance, testing, offsite support, and other features that the post-September 2001 regulations required. In March 2012, the NRC staff completed its review of licensee responses to Bulletin 2011-01. JLD-ISG-12-01, "Interim Staff Guidance for Compliance with Order EA-12-049 Concerning Mitigation Strategies," issued September 2012 addressed minor discrepancies identified during these reviews. This ISG provides guidance for compliance on license requirements for mitigation strategies for beyond-design-basis external events.

The NRC staff engaged stakeholders during public meetings to discuss the technical basis and acceptance criteria for conducting a reevaluation of site specific seismic hazards and to inform

NRC's process for defining guidelines for applying present-day regulatory guidance and methodologies being used for early site permits and combined license reviews for new reactors to the reevaluation of seismic and flooding hazards at operating reactors.

On March 12, 2012, and as discussed in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012, the NRC issued RFIs asking licensees to:

- reevaluate site-specific seismic and flooding hazards
- identify actions taken or actions planned to address plant-specific issues associated with the updated seismic and flooding hazards
- identify and address plant-specific issues and verify the adequacy of monitoring and maintenance for protection features by performing seismic and flooding walkdowns
- inform the NRC of the results of the walkdowns and corrective actions taken or planned

Licensees completed the seismic and flooding walkdowns and submitted reports to the NRC in November 2012. NRC inspectors accompanied and independently verified that the walkdowns were performed in accordance with JLD-ISG-12-04, "Interim Staff Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter," and JLD-ISG-05, "Interim Staff Guidance on Performance of an Integrated Assessment for Flooding," both issued on November 2012. Identified discrepancies were entered into licensee corrective action programs.

While undertaking actions to address the seismic and flooding hazards discussed above, the NRC staff recognized that it should reevaluate other external hazards against existing requirements and regulatory guidance. Other external hazards include phenomena such as tornados, hurricanes, severe winds, extreme temperatures, extreme precipitation, dust storms, forest fires, and volcanic activity. Thus, NTTF Recommendation 2.1 will be expanded to include consideration of other external hazards. The other external hazards evaluation will require significant resources from licensees and the NRC, as well as specialized expertise to review licensee reevaluations and to document results of NRC evaluations. Since sufficient resource flexibility, including availability of critical skill sets, does not currently exist, the staff prioritized the other external hazards evaluation of NTTF Recommendation 2.1 as a Tier 2 activity.

The NRC staff has engaged with stakeholders during public meetings to discuss its planned regulatory actions with regard to other external hazards. Once sufficient expertise and resources are available, the staff will undertake the following regulatory activities:

- identify the acceptance criteria and methodology for conducting a reevaluation of site specific external natural hazards
- issue RFIs to reevaluate other site-specific natural hazards
- identify actions that have been taken, or are planned, to address plant-specific issues associated with the updated natural hazards
- evaluate licensee responses and take appropriate regulatory action to resolve issues associated with updated site-specific natural hazards

The NRC staff plans to develop and issue the RFIs on the reanalysis for other external hazards 6 months after sufficient expertise and resources are available. The staff has not yet determined a schedule to evaluate the licensee responses to the RFIs. Based on the results of these evaluations, the NRC will decide whether additional regulatory actions will be needed. If applicable, the NRC will conduct its inspection activities subsequent to the issuance and implementation of these actions.

18.5.2 State-of-the-Art Analysis

Driven by technical advances since the last NRC-sponsored Level 3 PRAs were performed, the NRC staff is developing plans for a new Level 3 PRA. The last Level 3 PRA, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," was issued more than 20 years ago, in December 1990. Technical advances since that time include: (1) plant modifications to enhance nuclear power plant operational performance, safety, and security; (2) improved understanding and modeling of severe accident phenomena; and (3) advances in PRA technology, such as common-cause modeling.

The staff also has identified additional scope considerations that could be addressed in a new and more comprehensive Level 3 PRA. These factors include (1) multi-unit site effects; (2) other site radiological sources (e.g., SFPs or dry storage casks); and (3) site-specific external hazards such as fires, flooding, and seismic events.

A new full-scope comprehensive site Level 3 PRA that incorporates these technical advances and additional scope considerations could improve the NRC's understanding of probable risk, enhancing regulatory decisionmaking, and helping the agency focus its limited resources on issues most pertinent to its mission to protect public health and safety. The full description of Level 3 PRA activities is discussed in SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities," dated July 7, 2011, and the associated SRM, dated September 21, 2011. In SECY-11-0089, the staff presented three options to the Commission for performing a Level 3 PRA: (1) maintain the status quo and continue evolutionary development of PRA technology, (2) conduct focused research to address identified gaps in existing PRA technology before performing a full-scope comprehensive site Level 3 PRA, and (3) conduct a full-scope comprehensive site Level 3 PRA. In the SRM response, the Commission approved a modified version of option 3 to conduct a full-scope comprehensive site Level 3 PRA for an operating plant. The modification to option 3 extended the schedule to 4 years to alleviate some of the near-term resource challenges and allow adequate time for a careful site selection process. The new Level 3 PRA will offer insight into many of the NTTF recommendations.

18.5.3 Events Beyond the Current Design-Basis

The staff's review of the NTTF recommendations identified areas for further evaluation to enhance the regulations and cope with events beyond the current design-basis. Essentially all of the actions the NRC is pursuing relate to events beyond the current design-basis.

In addition to the activities discussed in previous Articles of this report, the NRC is evaluating additional topics related to external events beyond the design-basis. Evaluations are underway assessing the possible merits of and basis for:

- a requirement that licensees confirm seismic and flooding hazards every 10 years and address any new and significant information
- potential enhancements to licensees' capabilities to prevent or mitigate seismically induced fires and floods

The NRC plans for performing work on these topics are discussed in SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012.

ARTICLE 19. OPERATION

Each Contracting Party shall take appropriate steps to ensure that:

(i) the initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning program demonstrating that the installation, as constructed, is consistent with design and safety requirements

(ii) operational limits and conditions derived from the safety analysis, test, and operational experience are defined and revised as necessary for identifying safe boundaries for operation

(iii) operation, maintenance, inspection, and testing of a nuclear installation are conducted in accordance with approved procedures

(iv) procedures are established for responding to anticipated operational occurrences and to accidents

(v) necessary engineering and technical support in all safety related fields is available throughout the lifetime of a nuclear installation

(vi) incidents significant to safety are reported in a timely manner by the holder of the relevant license to the regulatory body

(vii) programs to collect and analyze operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies

(viii) the generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal

The NRC relies on regulations in 10 CFR and internally developed associated programs in granting the initial authorization to operate a nuclear installation and in monitoring its safe operation throughout its life. This section describes the most significant regulations and programs corresponding to each obligation of Article 19. It also includes a discussion on lessons learned from Fukushima.

19.1 Initial Authorization to Operate

All currently operating reactors in the United States received licenses under the two-step process in 10 CFR Part 50. This licensing process requires both a construction permit and an operating license. The additional licensing processes in 10 CFR Part 52 provide for site approvals and design approvals in advance of construction authorization. In addition, 10 CFR Part 52 includes a process that combines a construction permit and an operating license with conditions into one license (a combined license). Both the two-step and the combined license processes require NRC approval to construct and operate a nuclear power plant.

The Advisory Committee on Reactor Safeguards, an independent statutory committee established to advise the NRC on reactor safety, reviews each application to construct or operate a nuclear power plant. The committee begins its review early in the licensing process by selecting the proper stages at which to meet with the applicant and NRC staff. Upon completing its review, the committee reports to the Commission.

The public also has an opportunity to have its concerns addressed. The Atomic Energy Act and the NRC's regulations implementing this Act require the NRC to hold a public hearing before it may issue a construction permit, early site permit, or combined license for a nuclear power plant. Three-member Atomic Safety and Licensing Boards, which consist of one legal judge who acts as the chairperson and two technically qualified judges from the Atomic Safety and Licensing Board Panel, conduct public hearings for applications for construction permits and early site permits. For combined licenses, the Commission conducts the uncontested mandatory public hearing, while Atomic Safety and Licensing Boards conduct any contested hearings on these license applications if a request for such a hearing is filed and granted. Members of the public may submit written statements as part of these hearings, or they may petition for leave to intervene as full parties in the hearing.

To obtain NRC approval to construct or operate a nuclear power plant, an applicant must submit safety analysis and environmental reports. Article 18 describes the final safety analysis report and the NRC's review of the application for an operating license. Unlike the process for an application for a construction permit, early site permit, or combined license, a public hearing is neither mandatory nor automatic for an application for an operating license under 10 CFR Part 50. However, soon after the NRC accepts the application for review, it publishes a notice in the *Federal Register* stating that it is considering issuing the license. This notice states that any person whose interest might be affected by the proceeding may petition the NRC for a hearing. Similar to the public hearings on applications for construction and early site permits, three-member Atomic Safety and Licensing Boards conduct any public hearings on applications for operating licenses. A licensing board will also determine whether to grant or deny the request for a hearing.

An early site permit issued under Subpart A, "Early Site Permits," to 10 CFR Part 52, provides for resolution of site safety, environmental protection, and emergency preparedness issues, independent of a specific nuclear plant design review. The application for an early site permit must address the safety and environmental characteristics of the site and evaluate potential physical impediments to the development of an acceptable emergency plan or security plan. The applicant may submit additional information on emergency preparedness issues up to a complete emergency plan. The staff documents its findings on site safety characteristics and emergency planning in a safety evaluation report and its findings on environmental protection issues in an environmental impact statement. The early site permit may also allow limited construction activities in accordance with 10 CFR 50.10, "License Required; Limited Work Authorization," subject to redress, before the issuance of a combined license. The NRC will issue a *Federal Register* notice for a mandatory public hearing, and the Advisory Committee on Reactor Safeguards will perform an independent safety review. The duration of an early site permit is 10 – 20 years, and the permit may be renewed. A construction permit or combined license application may reference the early site permit.

The NRC also may certify a standard plant design through a rulemaking under Subpart B, "Standard Design Certifications," to 10 CFR Part 52. The design certification process resolves final design information for an essentially complete plant, independent of a specific site, and the

Advisory Committee on Reactor Safeguards performs an independent safety review. The NRC has certified four standard plant designs under the design certification process: General Electric's ABWR, and Westinghouse's System 80+ (originally designed by Combustion Engineering), AP600, and AP1000. The duration of a design certification is 15 years, and the certification may be renewed.

A combined license, issued under Subpart C, "Combined Licenses," to 10 CFR Part 52 authorizes construction of a facility in a manner similar to a construction permit under 10 CFR Part 50. An application for a combined license may incorporate by reference an early site permit, design certification, both, or neither. The advantage of referencing an early site permit or design certification is that issues resolved during those processes are not considered again at the combined license stage. Just as for a construction permit, the NRC must hold a hearing before deciding whether to issue a combined license. However, the combined license will specify the inspections, tests, and analyses that the licensee must perform and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the applicable regulations.

After issuing a combined license, the NRC staff will verify that the licensee has performed the required inspections, tests, and analyses, and before operation of the facility the Commission must find whether the licensee has met the acceptance criteria. The licensee must submit notifications to the NRC during construction that it has successfully performed the inspections, tests, and analyses, and met the acceptance criteria. Periodically during construction, the NRC staff will publish notices of the successful completion of inspections, tests, and analyses in the *Federal Register*. Not less than 180 days before the date scheduled for initial loading of fuel, the NRC will publish a notice of intended operation of the facility in the *Federal Register*. Affected members of the public have an opportunity to request a hearing on whether the facility complies or will comply with the accepted criteria. However, requests for such a hearing will be considered only if the petitioner demonstrates that one or more of the acceptance criteria have not been (or will not be) met, and the specific operational consequences of nonconformance would be contrary to providing reasonable assurance of adequate protection of the public health and safety.

19.2 Definition and Revision of Operational Limits and Conditions

The license for each nuclear facility must contain technical specifications that set operational limits and conditions derived from the safety analyses, tests, and operational experience. The regulations contained in 10 CFR 50.36 define the requirements that apply to the plant-specific technical specifications. At a minimum, the technical specifications must describe the specific characteristics of the facility and the conditions for its operation that are required to adequately protect the health and safety of the public. Each applicant must note items that directly apply to maintaining the integrity of the physical barriers designed to contain radioactive material. In 10 CFR 50.36, the NRC requires that the technical specifications must be derived from the analyses and evaluations in the safety analysis report. Licensees cannot change the technical specifications without prior NRC approval.

In 1992, the NRC issued improved, vendor-specific (e.g., Babcock & Wilcox, Westinghouse, Combustion Engineering, and General Electric) standard technical specifications in NUREGs 1430-1434 and periodically revises them on the basis of experience. The NRC issued Revision 4 to these NUREGs in April 2012.

The NRC encourages licensees to use the improved standard technical specifications as the basis for plant-specific technical specifications. The agency also considers requests to adopt parts of the improved standard technical specifications, even if the licensee does not adopt all of the improvements. These parts, which will include all related requirements, will normally be developed as line-item improvements. To date, over half of the operating commercial nuclear plants have converted their technical specifications to the improved standard technical specifications.

Consistent with the Commission's policy statements on technical specifications and the use of PRAs, the NRC and the nuclear industry are developing risk-informed improvements to technical specifications. These improvements and initiatives are intended to maintain or improve safety while reducing unnecessary burden and to make technical specifications congruent with the agency's other risk-informed regulatory requirements (in particular, the risk management requirements of the Maintenance Rule in 10 CFR 50.65(a)(4)).

19.3 Approved Procedures

In the United States, operations, maintenance, inspection, and testing of a nuclear installation are conducted in accordance with approved procedures. Each nuclear facility is required to follow the quality assurance requirements in Appendix B to 10 CFR Part 50. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, requires that licensees establish measures to ensure that activities that affect quality will be prescribed by appropriate documented instructions, procedures, or drawings. RG 1.33, Revision 2, provides supplemental guidance.

19.4 Procedures for Responding to Anticipated Operational Occurrences and Accidents

The NRC has provided guidance on responding to anticipated operational occurrences and accidents in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980; NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," issued January 1983; and NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," issued August 1982.

After the 1979 accident at Three Mile Island Unit 2, the NRC issued orders requiring licensees to develop procedures for coping with certain plant transients and postulated accidents. It also issued NUREG-0737 in 1980 and Supplement 1 to that document in 1983, which recommend that licensees develop procedures to cope with accidents and transients that are caused by initiating events analyzed in the final safety analysis report with multiple failures of equipment.

NUREG-0899 gives programmatic guidance for developing emergency operating procedures. To ensure that proper procedures had been developed to respond to plant transients and accidents, the NRC reviewed each plant using the guidance in NUREG-0800, Section 13.5.2.1. Section 1.3.2 of this report provides details on the tiered recommendations from the NRC's Fukushima NTTF.

19.5 Availability of Engineering and Technical Support

The NRC's Reactor Oversight Process, described in Article 6 of this report, includes techniques to ensure that adequate engineering and technical support is available throughout the lifetime of a nuclear installation. Several of the IPs focus on ensuring the maintenance of adequate support programs. Licensees also report performance indicators. Depending on inspection findings and performance indicators, the NRC conducts additional inspections to focus on the causes of the performance problems as prescribed by the Reactor Oversight Process Action Matrix.

19.6 Incident Reporting

Two of the many elements contributing to the safety of nuclear power plants are emergency response and the feedback of operating experience into plant operations. The licensee event reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 help to achieve these goals, as 10 CFR 50.72 requires immediate notification requirements through the emergency notification system, and 10 CFR 50.73 requires 60-day written licensee event reports. All 10 CFR 50.72 event notifications and 10 CFR 50.73 licensee event reports, except those containing sensitive security-related information, are publicly available on the NRC Web site.

The NRC staff uses the information reported under these regulations to respond to emergencies, monitor ongoing events, confirm licensing bases, study potentially generic safety problems, assess trends and patterns of operational experience, monitor performance, identify precursors of more significant events, and provide operational experience to the industry. Evaluations of events as documented in NRC inspection reports are publicly available on the NRC Web site. The annual abnormal occurrence report to Congress (NUREG-0090, “Report to Congress on Abnormal Occurrences”), which details specific events that result in a conditional core damage probability greater than 1×10^{-4} and other events of significant interest, is also publicly available.

The NRC modified these rules in 1992 and 2000. The modified rules continue to provide the Commission with reports of significant events for which the NRC may need to act to maintain or improve reactor safety, or to respond to heightened public concern. The modified rules also better align requirements on event reporting with the type of information that the NRC needs to carry out its safety mission. The NRC issued NUREG-1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” Revision 2, in October 2000, concurrent with the rule changes. NUREG-1022, Revision 3, published in January 2013 and effective July 2013, revises the event reporting guidelines in NUREG-1022, Revision 2, to provide clearer guidance.

NUREG-1022 is structured to help licensees promptly and completely report specified events and conditions. It discusses general issues that have been difficult to implement in the past, such as engineering judgment, time limits for reporting, multiple failures and related events, deficiencies discovered during licensee engineering reviews, and human performance issues. It also includes a comprehensive discussion of each reporting criterion with illustrative examples and definitions of key terms and phrases.

Event reporting under these rules since 1984 has contributed significantly to focusing the attention of the NRC and the nuclear industry on the lessons learned from operating experience to improve reactor safety. Over the years, improvements in reactor safety system performance and decreasing trends in the number of reactor transients and significant events have been evident. Between 2007 and 2012, there were no significant U.S. reactor events (defined as having a conditional core damage probability greater than 1×10^{-4}).

The NRC reviews each reported reactor-related event and assigns a rating of 1 through 7 or below scale on the International Nuclear and Radiological Event Scale. The agency submits events with a rating of 2 or higher to the IAEA nuclear events Web-based system for public posting. Other events that attract international public interest are also considered for posting regardless of the International Nuclear and Radiological Event Scale rating. The NRC describes this process in RIS 2002-01, “Changes to NRC Participation in the International Nuclear Event Scale,” issued January 2002, and IN 2009-27, “Revised International Nuclear and Radiological Event Scale User’s Manual,” issued November 2009.

19.7 Programs To Collect and Analyze Operating Experience

As outlined in GL 82-04, “Use of INPO SEE-IN (Significant Event Evaluation and Information Network) Program,” issued March 1982, INPO and the individual licensees are jointly responsible for compiling and analyzing operating experience within the industry. In November 2011, INPO replaced the Significant Event Evaluation and Information Network program with the Operating Experience and Construction Experience programs. These programs use four different levels of INPO event reports to communicate significant events to the industry. In addition, INPO’s Consolidated Events System provides member utilities with the ability to report lower level events and equipment failure data to INPO. The data is shared with all INPO members and, in a limited fashion, with the NRC.

The NRC Operating Experience Program consists of a process with four phases: (1) collection, (2) screening, (3) evaluation, and (4) application of operating experience data, with a common theme of communication running throughout.

The NRC facilitates the collection, storage, and retrieval of operating experience data with the Operating Experience Gateway, a centralized repository of links to databases relevant to operating experience on the NRC internal Web site, including event reports, international reports, and inspection findings. Since 2010, a broader database has been providing the same type of centralized data storage and retrieval options for lower level operating experience, which can be a useful source of information for long-term trending and analysis even when the issues do not rise to the threshold of reportable events.

The NRC reviews event notifications and lower level operating experience from resident inspector feedback to the regional offices daily to determine the level of followup each item requires. The NRC also considers licensee event reports, reports of defects and noncompliance submitted under 10 CFR Part 21, “Reporting of Defects and Noncompliance,” international operating experience received from the International Nuclear and Radiological Event Scale Web site and from the IAEA International Reporting System for Operating Experience, and any items of potential interest brought forward by the Office of New Reactors and the Office of Nuclear Regulatory Research.

Items that do not require significant evaluation are still reviewed and considered for followup actions. These can include email notification of technical staff review for event analysis and trending or an operating experience communication distributed internally throughout the agency summarizing the issue and its safety significance. Items that meet the criteria for both safety significance and generic applicability are held for further evaluation. This evaluation will generally involve an in depth examination of the technical aspects of each issue, its potential safety significance, and a review of previous operating experience.

Finally, the operating experience program applies the results of these evaluations. An operating experience application may include the issuance of a generic communication, a proposal for rulemaking, a referral for further study as a GSI, or a revision of IPs.

The NRC's construction experience program is described in Section 18.4 of this report. The NRC established a Cyber Assessment Team in 2009 to assess and provide analysis of cyber security related issues and events that may warrant NRC action and to provide technical support during NRC's response to events that may have cyber security implications.

The NRC participates in the International Nuclear and Radiological Event Scale and the IAEA incident reporting system to both communicate operating experience internationally and review events that other member States have posted. Operating experience personnel review all reactor event notifications the agency receives and rates them on the International Nuclear and Radiological Event Scale. As Section 19.6 of this report discusses, events with a rating of 2 or higher are posted to the International Nuclear and Radiological Event Scale Web site within 48 hours. The NRC screens all international events posted to this Web site to determine the appropriate level of evaluation required based on safety significance and applicability to U.S. plants. The NRC uses the same criteria to screen the IAEA's incident reporting system reports as they are posted. The NRC submits all U.S. reactor-related generic communications to the IAEA incident reporting system for communication to the international community along with selected licensee event reports related to events that have attracted international interest.

19.8 Radioactive Waste

The NRC has regulations and guidance for nuclear power reactor licensees to ensure the safe management and disposal of low-level radioactive waste. Onsite low-level waste must be managed in accordance with the NRC regulations in 10 CFR Part 20 and 10 CFR Part 50. For example, Subpart K, "Waste Disposal," to 10 CFR Part 20, deals with licensee treatment and disposition of radioactive waste. In addition, GL 1981-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," dated November 10, 1981, provides guidance on measures for ensuring the safe storage of low-level waste. The low-level waste storage guidelines were last updated in RIS 2011-09, "Available Resources Associated with Extended Storage of Low-Level Waste" in August 2011.

Notwithstanding these regulations and guidance, the economics of waste disposal in the United States have encouraged practices to minimize radioactive waste. In the past decade or so, disposal costs have risen significantly, and volumes of waste produced have decreased greatly as operations technology evolves. In June 2008, the NRC published RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning." Additionally, in May 2012, the NRC published the Policy Statement on Low-Level Radioactive Waste Management and Volume Reduction. The Policy Statement is a revision of the NRC's 1981 Policy Statement on Low-Level Radioactive Waste Volume Reduction to encourage licensees to take steps to reduce the amount of waste generated and to reduce the volume of waste once generated. Currently, nuclear power reactors generate only small amounts (about 1,000-2,000 cubic feet per unit) of operational waste each year.

For storage, waste is conditioned into a form that is stable and safe to minimize the likelihood that it will migrate. Waste placed into storage is in a form that is suitable for disposal, or at least a form that can be made suitable for future disposal. The NRC maintains specific regulations for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related low-level waste greater than Class C[10] in 10 CFR Part 72 and detailed regulations for designing and operating low-level waste disposal facilities in 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste.”

The U.S. Government addresses in detail the spent fuel and radioactive waste programs, including high-level waste, in a report prepared to satisfy the reporting requirements of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The latest report (DOE/EM-0654, “United States of America Fourth National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management,” Revision 3, issued September 2011) is available on the DOE Environmental Management Web site. In January 2013, DOE released the Administration’s “Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste.” The strategy endorses a waste management system containing a pilot interim storage facility; a larger, full-scale interim storage facility; and a geologic repository in a timeframe that demonstrates the Federal commitment to addressing the nuclear waste issue, builds capability to implement a program to meet that commitment, and prioritizes the acceptance of fuel from shutdown reactors. DOE’s proposed strategy was developed in response to the final recommendations that the Blue Ribbon Commission on America’s Nuclear Future provided to the U.S. Secretary of Energy in January 2012. The NRC will continue to ensure the safe storage of civilian high-level waste.

Details on the status of the Waste Confidence Decision and Rule are discussed in Sections 6.3.9 and 14.1.4.1 of this report.

19.9 Fukushima Lessons Learned

Immediately following the accident at Fukushima in Japan, the NRC took actions that verified nuclear power plant operators’ preparedness to respond to and mitigate the consequences of beyond-design-basis events. These actions included the issuance of IN 2011-05, “Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants,” dated March 18, 2011, and IN 2011-08, “Tohoku-Taiheiyou-Oki Earthquake Effects On Japanese Nuclear Power Plants – For Fuel Cycle Facilities,” dated March 31, 2011, to inform U.S. licensees regarding what was known about the Fukushima accident.

The NRC also issued TI 2515/183, and TI 2515/184, “Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs),” dated April 29, 2011, to NRC regional and resident inspectors to evaluate specific aspects of licensee preparedness to respond to an event like that which occurred at the Fukushima facility. The inspections under TI 2515/183 focused on independently assessing the adequacy of licensee equipment, capabilities, and strategies to respond to large fires, explosions, SBO events, and flooding. NRC inspectors walked down the facilities and looked at operability of equipment, reviewed procedures, looked at maintenance, and reviewed training programs. The inspections under TI 2515/184 focused on evaluating the adequacy of the SAMGs and how they have been maintained over the years.

Finally, the NRC issued Bulletin 2011-01, to request information from U.S. licensees regarding mitigating strategies for large fires and explosions and to confirm what was put into place

10 NRCs classification system contained in 10 CFR Part 61 includes Class A, B, and C low level waste that is suitable for land disposal. Low level waste that does not meet the criteria for these classes is considered greater than Class C and eventually will be managed by DOE in a yet-to-be-determined manner. Until then, such waste must be managed (stored) by licensees. Regulations in 10 CFR Part 72 allow, but do not require, the onsite management of greater than class C low level waste in independent storage facilities separate from the ones used to manage spent fuel.

following the events of September 11, 2001. By mid-June 2011, licensees were required to verify that equipment and staff were capable of performing the required actions. By mid-July 2011, licensees reported to the NRC on the equipment maintenance and testing, offsite support, and other features that the post-September 2001 regulations required.

In parallel, on March 23, 2011, the Commission directed the staff to conduct a review of the lessons learned from the Fukushima accident. The staff was directed to review all available information on the event, evaluate the effectiveness of NRC regulatory processes in this area, and recommend near-term actions and long-term activities within 90 days. The near-term review was conducted by a task force (i.e., NTTF) of senior NRC managers and staff. The NTTF completed its review and issued "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near Term Task Force Review of the Insights from the Fukushima Dai-ichi Accident," dated July 12, 2011, providing recommendations to the Commission.

The NTTF concluded that given the NRC's current regulatory approach, and more importantly, the resultant plant capabilities, a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented to reduce the likelihood of core damage and radiological releases. Therefore, continued operation and continued licensing activities do not pose an imminent risk to public health and safety. However, the NTTF also concluded that enhancements to safety and emergency preparedness were warranted, and it made 12 overarching recommendations for consideration. Details on these recommendations are discussed in Part 1 of this report.

The NRC also began a SFP study, which considered a SFP similar to the one at Fukushima and 23 U.S. reactors, and an earthquake several times stronger than what the SFP's design considered. The study examined both a full SFP and one with less fuel and more spacing between individual fuel assemblies, as well as emergency procedures for adding water to the pool in the unlikely event that the earthquake causes the pool to lose water. The detailed analysis showed that even a very strong earthquake has a low probability of damaging the pool to the point of losing water. The draft study also showed that even if this particular pool was damaged, the fuel could be kept cool in all but a few exceptional circumstances.

In cases where the analysis led to fuel damage, the draft study concluded that existing emergency procedures would keep the population around the plant safe. Those emergency measures could mean relocating people from a large area of potentially contaminated land. The study also examined the potential benefits of moving all spent fuel older than 5 years (and therefore easier to cool) into storage casks. For the scenarios examined, the study concluded faster fuel transfer to casks would not provide a significant safety benefit for the plant. The NRC will use the final study to inform further analysis of SFPs in the U.S.

PART 3

Convention on Nuclear Safety Report:

The Role of the Institute of Nuclear Power Operations in Supporting the United States Commercial Nuclear Power Industry's Focus on Nuclear Safety

August 2013

INPO®

1. Executive Summary

Following the event at Three Mile Island Nuclear Station (Three Mile Island), the U.S. nuclear power industry established the Institute of Nuclear Power Operations (INPO or "the institute") in 1979 to promote the highest levels of safety and reliability (i.e., to promote excellence) in the operation of its nuclear power plants. INPO is a nongovernmental corporation that operates on a not-for-profit basis. Under the United States (U.S.) tax law, the company is classified as a charitable organization that "relieves the burden of Government."

Since its inception, all organizations that have direct responsibility and legal authority to operate or construct commercial nuclear plants in the United States have maintained continuous membership in INPO, which currently has 25 members. In addition, many organizations that jointly own these nuclear power plants are associate members. A number of international utility organizations and major supplier organizations also voluntarily participate in the institute's activities and programs.

In forming INPO, the nuclear utility industry took an unusual step. The industry placed itself in the role of overseeing INPO activities while endowing INPO with ample authority to bring pressure for change on individual members and the industry as a whole. This feature makes INPO unique. The industry clearly established and accepted a form of self-regulation through peer review by helping to develop INPO performance objectives and criteria (POCs) and then by committing to meet these POCs. The industry's recognition that all nuclear utilities are affected by the action of any one utility motivated its support of INPO. Each individual member is solely responsible for the safe operation of its nuclear plants. The U.S. Nuclear Regulatory Commission (NRC) has statutory responsibility for overseeing the licensees and for verifying that each licensee operates its facility in compliance with Federal regulations to ensure public health and safety. INPO's role -- encouraging the pursuit of excellence in the operation of commercial nuclear power plants -- is complementary but separate and distinct from the role of the NRC.

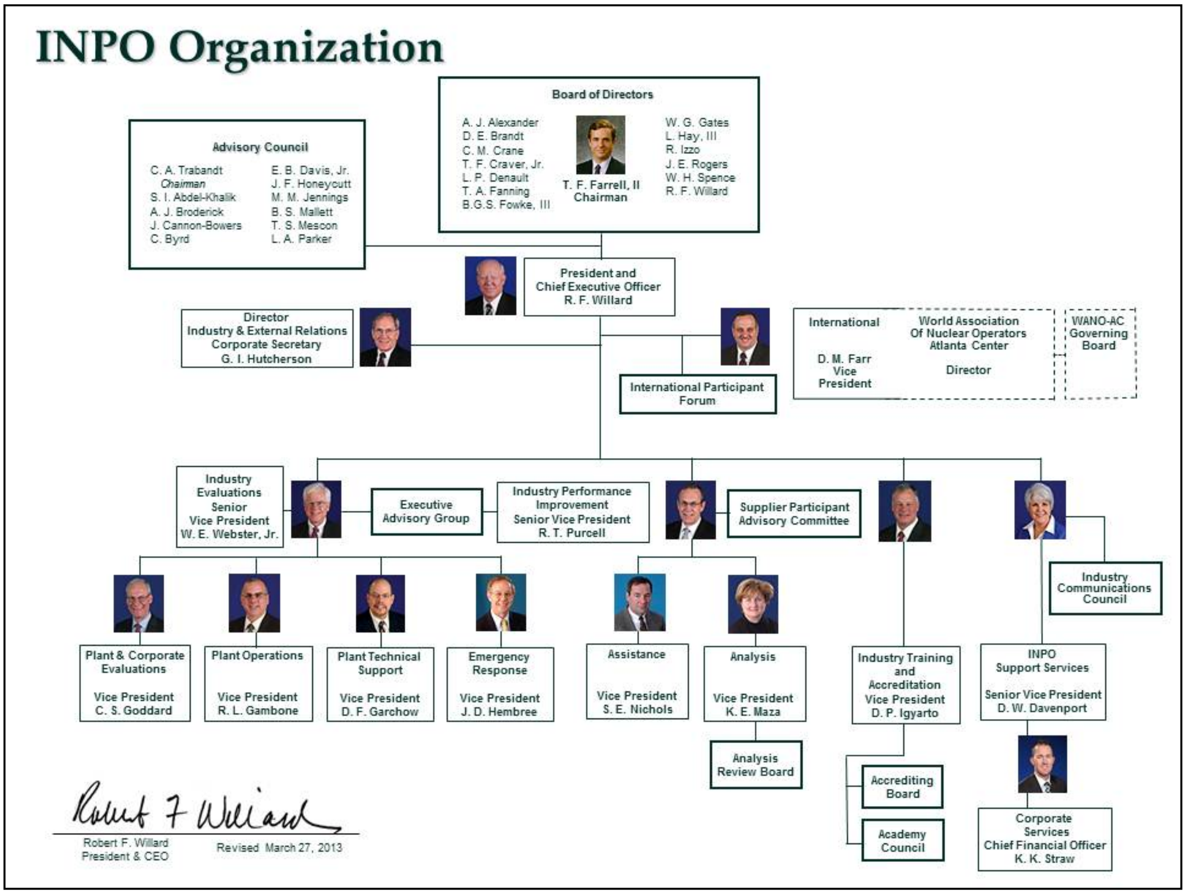
The nuclear industry's commitment to go beyond regulatory compliance and continually strive for excellence, with INPO's support, has resulted in substantial performance improvements over the last 30 years. For example, in the early 1980s the typical nuclear plant had a capacity factor of 63 percent, had experienced six automatic scrams per year, had high collective radiation dose, and had experienced numerous industrial safety accidents among its staff. Today, the median industry capacity factor is above 91 percent, most plants have no automatic scrams per year, and collective radiation dose and industrial accident rates are both lower by a factor of 7 when compared to the rates of the 1980s.

The earthquake and tsunami in Japan on March 11, 2011, and subsequent nuclear accident at Tokyo Electric Power Company's Fukushima Dai-ichi nuclear power plant have resulted in worldwide attention toward improving nuclear safety.

One of the intents of this report is to provide an understanding of the institute's role and its major programs in response to the accident at Fukushima.

2. Organization and Governance

In many ways, INPO's organizational structure is similar to that of a typical U.S. corporation. A Board of Directors, comprising of senior executives from INPO's member organizations, provides overall direction for the institute's operations and activities. Currently, the Board of Directors comprises 13 chief executive officers (CEOs) and one president from the member utilities. The institute's bylaws specify that at least two directors must have recent experience in the direct supervision of the operation of a facility that generates electricity or steam for commercial purposes through the application of nuclear power. In addition, at least one director must represent a public utility. The president and CEO of the institute, normally a single individual, is elected by, and reports to, the Board of Directors. INPO's organization chart is presented below.



Because the INPO Board of Directors is made up of utility executives, the industry believes that having support from an Advisory Council of distinguished individuals, mainly from outside the nuclear generation industry, to provide diversity of experience and thought is also important. This Advisory Council of 9 to 15 professionals selected from outside INPO's membership meets periodically to review the institute's activities and to provide advice on broad objectives and methods to the Board of Directors. Members include prominent educators; scientists; engineers; business executives; and experts in organizational effectiveness, human relations, and finance.

The institute's activities to enhance nuclear plant safety and reliability are reflected primarily in its four cornerstone programs: (1) periodic onsite evaluations of each nuclear plant and corporate support organizations, (2) training and accreditation, (3) events analysis and information exchange, and (4) assistance. INPO has organized nuclear technical divisions to carry out the cornerstone functions. Other functional areas, such as support services, industry and external relations, and communications, support INPO's nuclear technical divisions and its overall mission.

The National Academy for Nuclear Training (hereafter referred to as "the Academy") operates under the direction of INPO and integrates the training efforts of all U.S. nuclear utilities, the activities of the National Nuclear Accrediting Board, and the training-related activities of the institute. An INPO executive serves as the executive director of the Academy.

Non-U.S. nuclear organizations from 18 different countries or provinces participate in the INPO's International Participant Program and are managed by the World Association of Nuclear Operators (WANO)-Atlanta Centre at the institute's request. This program involves the active exchange of information on nuclear plant operations among utility organizations around the world. Each international participant organization is represented on an advisory committee that provides advice on the operation of this program and input on other INPO programs, as appropriate. An INPO executive serves as the director of the WANO – Atlanta Centre.

Organizations engaged in providing commercial design, engineering, nuclear fuel cycle, or other services directly related to the construction, operation, or support of nuclear electric generating plants also participate in INPO through the Supplier Participant Program. This program allows supplier organizations to share experience and expertise with INPO members and provides a way to give feedback on operational experience to the suppliers. Currently, 26 companies from around the world are involved in the Supplier Participant Program.

The industry actively participates in the oversight of INPO's programs. Representatives from member utilities serve on the Executive Advisory Group, the Academy Council, the Analysis Review Board, and the Industry Communications Council. The Executive Advisory Group, which comprises of the chief nuclear officers of all the member organizations, advises INPO management on the programs and products in the nuclear technical areas. The Academy Council provides advice in the areas of training, accreditation, and human performance. The Analysis Review Board advises INPO on analysis activities, and the Industry Communications Council advises on effective communication of INPO programs and activities. Frequently, INPO establishes ad hoc industry groups to provide input on specific initiatives.

Financial and Human Resources

The 2012 operating budget for INPO was $114 million, which is primarily funded through member dues. Dues are approved annually by the Board of Directors and are assessed based on the number of each member's nuclear plant sites and units.

INPO's permanent staff of about 340 is augmented extensively by industry professionals who serve as loaned employees or international liaison engineers on assignments of typically 18 to 24 months. Loaned and liaison employees comprise about one-third of the total technical staff. They gain extensive experience and training while providing current industry expertise and diversity of thought and practices. A small number of permanent INPO employees serve in loaned assignments to member organizations, primarily for professional development. The total number of both permanent and loaned employees is approximately 400 people.

INPO resources and capabilities are further enhanced by the extensive use of U.S. and international utility peers and executive industry advisors. These peers participate in a wide range of short-term activities, especially on evaluation and accreditation teams that visit nuclear plants. Peers enhance the effectiveness of the INPO teams by offering varied perspectives and by providing additional current experience. The peers benefit from learning other ways to conduct business that can be shared with their stations. In 2012, the industry provided INPO with more than 650 peers for short term assignments.

3. INPO's Role within the Federal Regulatory Framework

The Federal Government regulates the nuclear utility industry in the United States, as it does other industries that may affect the health and safety of the general public. This regulatory function is based principally on the Atomic Energy Act of 1954, as amended, and is carried out by the NRC. In 1979, following the accident at Three Mile Island, the President of the United States appointed a commission to investigate the accident. The commission, which came to be known as the Kemeny Commission, helped influence the industry's decision to create INPO as a method of self-regulation.

The industry created INPO to provide the means whereby the industry itself could, acting collectively, improve the safety and reliability of nuclear operations. Industry leaders envisioned that peer reviews and POCs based on excellence would be effective in bringing about improvements. In the broad sense, the ultimate goals of the NRC and INPO are the same in that both organizations strive to protect the public; therefore, both review similar areas of nuclear power plant operations. In granting INPO its not-for-profit status, the U.S. Government acknowledged that INPO's role reduces the burden on the Government through the conduct of its activities. However, the industry does not expect INPO to supplant the regulatory role of the NRC. INPO recognized that in establishing and meeting its role, it would have to work closely with the NRC while not becoming or appearing to become an extension of, or an advisor to, the NRC or an advocacy agent for the utilities. As recognition of their different roles but common goals, the NRC and INPO have entered into a memorandum of agreement that includes coordination plans that cover specific areas of mutual interest.

The conduct of plant and corporate evaluations is one of INPO's most important functions. It is also the function that is closest to the role of a regulator. Although the two roles -- evaluation and regulation -- may appear similar, they do differ in some ways. The industry and INPO jointly develop numerous POCs. INPO then conducts regular, extensive, and intrusive evaluations to determine how well they are being met. These POCs are broad statements of conditions that reflect a higher level of overall plant performance—striving for excellence and often exceeding regulatory requirements. These POCs, by their very nature, are difficult to achieve consistently.

Because of the differences in the roles of INPO and the NRC, the industry maintains a clear separation between INPO evaluations and NRC inspections. The industry expects INPO to keep the NRC apprised of its generic activities. Although INPO interactions with an individual member remain private between that member and INPO, stations are encouraged to make their INPO plant evaluation and accreditation results available to the NRC for review at each utility or site.

The industry recognizes the need for the NRC to assess the overall quality of INPO's products and the success of its programs. Therefore, the industry expects INPO to provide the NRC with information on INPO programs and activities, including the following:

- copies of selected generic documents
- access to other pertinent information, such as the Equipment Performance Information Exchange (EPIX) database, as described in specific agreements
- observation of certain INPO field activities by NRC employees, with agreement from members
- observation of National Nuclear Accrediting Board sessions

INPO regularly participates in industry-led working groups and task forces that interface with the NRC on specific regulatory issues and initiatives relative to the institute's mission and strategic objectives. These cooperative interactions have led to the elimination of some redundant activities, thus benefiting INPO members while enabling both the NRC and INPO to maintain or strengthen the focus on their respective missions. For example, the Consolidated Data Entry System, operated by INPO, collects operating data that the NRC uses in its industry oversight process.

INPO has implemented a policy and appropriate procedures on the handling of items that are potentially reportable to the NRC. INPO's policy is to inform utility management of such items during the normal course of business so that the utility can evaluate and report the items as appropriate. If INPO becomes aware of a defect or failure to comply that requires a report under Federal regulation, the institute has an obligation to ensure that the item is reported, if the utility has not already done so.

4. Responsibilities of INPO and Its Members

INPO members are expected to strive for excellence in the operation of their nuclear plants, to meet INPO POCs, and to meet the intent of INPO guidelines. This effort also includes the achievement and maintenance of accredited training programs for personnel who operate, maintain, and support their nuclear plants. Members are expected to be responsive to all areas for improvement identified through INPO evaluation, accreditation, and events analysis programs.

Nuclear operators are explicitly responsible for complying with the terms and conditions of the operating license and the applicable rules and regulations. The licensee is ultimately responsible for the safety of its activities and the safeguarding of nuclear facilities and materials used in operation. This concept is a key principle concerning INPO's relationship with its members.

The INPO Board of Directors approved a special procedure that provides guidance if a member is not responsive to INPO programs, if it is unwilling or unable to take action to resolve a significant safety issue, has persistent shortfalls in performance, or if the accreditation for its training programs has been put on probation or withdrawn by the National Nuclear Accrediting Board. The procedure specifies that INPO and the member's management work to resolve any issues in contention using a graduated approach of increasing accountability. Specific options for accountability include interactions between INPO's CEO and the member's CEO and, if necessary, its Board of Directors. One option also includes suspending INPO membership if the member continues to be unresponsive. Suspension of membership has never been necessary; however, such action would significantly affect the utility's continued operation, including limiting its ability to obtain insurance.

Furthermore, members are expected to participate fully in other generic INPO programs designed to enhance nuclear plant safety and reliability industrywide. Examples include providing INPO with detailed and timely operating experience information and participating fully in the loaned employee, peer evaluator, and WANO performance indicator programs. Members share information, practices, and experiences to assist each other in maintaining high levels of operational safety and reliability.

In return, the industry expects INPO to provide members with results from evaluation, accreditation, and review visits, including written reports and an overall numerical assessment that characterizes performance relative to standards of excellence. The industry expects INPO to followup on effective corrective actions by a member and to verify that the member has implemented these actions.

INPO and its members clearly understand that all parties must maintain the confidentiality of the institute's evaluation reports and related information and that members must not distribute this information external to their utility organizations. INPO also expects members and participants to use information provided by the institute to improve nuclear operations, not for other purposes (e.g., to gain commercial advantage). Members are to avoid involving INPO or INPO documents in litigation.

INPO members that are also members of the collective insurance organization, Nuclear Electric Insurance Limited (NEIL), have authorized and instructed INPO to make available to NEIL copies of its evaluation reports and other data at its office. NEIL reviews these reports and data for items that could affect the insurability of its members.

INPO POCs are written with input from, and with the support of, the industry. However, these POCs are written without regard to constraints or agreements, such as labor agreements, of any individual member. INPO expects each member to resolve any impediments to the implementation of the POCs that may be imposed by outside organizations.

INPO does not engage in public, media, or legislative activities to promote nuclear power. Such activities would undermine INPO's objectivity and credibility and may jeopardize the institute's not-for-profit status.

5. Principles of Sharing (Openness and Transparency)

Throughout the changes that have occurred in the U.S. electric industry, including the process of electric deregulation, the industry has reaffirmed INPO's mission to promote the highest levels of safety and reliability (i.e., to promote excellence) in the operation of nuclear power plants. Even with U.S. utilities now in competition in certain areas, these plants clearly understand the need to continue sharing pertinent operational information to continuously strengthen safety and reliability. Nuclear utility owners believe that this cooperation is fundamental to the industry's continued success.

Through INPO, nuclear utilities quickly share information important to safety and reliability, including operating experience, operational performance data, and information related to the failure of equipment that affects safety and reliability. The industry also actively encourages benchmarking visits to support the sharing of best practices and the concepts of emulation and continuous improvement.

INPO facilitates industry information sharing by including participation of industry peers in the institute's cornerstone programs—plant evaluations, training and accreditation, analysis and information exchange, and assistance. INPO communicates and shares information through a variety of methods, including the secure member Web site, Nuclear Network®, written guidelines, and other publications.

Although the industry and INPO recognize that the rapid and complete sharing of information important to nuclear safety is essential, both entities clearly understand that certain information is private in nature and is not appropriate to share. Examples are INPO plant-specific details of evaluation and accreditation results, personal employee and individual performance information, and appropriate cost and power marketing data.

6. Priority to Safety (Safety Culture)

The U.S. nuclear industry believes that a strong safety culture is central to excellence in nuclear plant operations, partly because of the special and unique nature of nuclear technology and the associated hazards—radioactive byproducts, concentration of energy in the reactor core, and decay heat. Within the INPO members' power plants and within INPO itself, the elements, activities, and behaviors that are part of a strong safety culture are embedded in everything that the institute does day-to-day and has been doing since its establishment in 1979.

The U.S. nuclear industry has defined safety culture as follows: An organization's values and behaviors—modeled by its leaders and internalized by its members—that serve to make nuclear safety the overriding priority.

On December 12, 2012, INPO distributed a report entitled "Traits of a Healthy Nuclear Safety Culture," and Addendum II to that report entitled, "Cross-References for Traits of a Healthy Nuclear Safety Culture." These documents were developed through a collaborative effort of the U. S. and international nuclear operating communities, representatives from NRC, the public, and INPO staff.

In July 2013, the report and its addendum will replace the INPO report "Principles for a Strong Nuclear Safety Culture," issued November 2004. To aid in the transition, Addendum II cross-references the new traits to the current safety culture principles, NRC safety culture language, and International Atomic Energy Agency (IAEA) safety culture attributes.

An additional addendum – Addendum I entitled "Behaviors and Actions that Support a Healthy Nuclear Safety Culture" is under development. Addendum I will include behaviors and examples, sorted by organizational level and attribute. This addendum is scheduled for distribution in early 2013. INPO will also provide pocket-sized copies of the traits and behaviors document in early 2013 and will make new safety culture posters available.

INPO activities reinforce the primary obligation of the operating organizations' leadership to establish and foster a healthy safety culture, to periodically assess safety culture, to address shortfalls in an open and candid fashion, and to ensure that everyone from the board room to the shop floor understands his or her role in safety culture.

As part of its focus on safety, the industry uses INPO, through evaluations and other INPO activities, to identify and help correct early signs of decline in the safety culture at any plant or utility. Furthermore, the industry has defined INPO's role doing the following:

- Define and publish standards relative to safety culture.
- Evaluate safety culture at each plant.
- Develop tools to promote and evaluate safety culture.
- Assist the industry in providing safety culture training.
- Develop and issue safety culture lessons learned and operating experience.
- Make safety culture visible in various forums such as professional development seminars, assistance visits, working meetings, and conferences, including the CEO conference.

In 2002, INPO published Significant Operating Experience Report (SOER) 02-4, "Reactor Pressure Vessel Head Degradation at Davis-Besse Nuclear Power Station." SOER 02-4 describes the event and the shortfalls in safety culture that contributed to it and recommends actions to prevent similar safety culture problems at other plants. The U.S. nuclear power industry considers this event a defining moment because it highlights problems that can develop when the safety culture at a plant receives insufficient attention. Every U.S. nuclear power station has implemented the recommendations in SOER 02-4, and INPO evaluation teams have reviewed each station's actions. Briefly, the recommendations encompass (1) discussing a case study on the event with all managers and supervisors in the nuclear organization, (2) periodically conducting a self-assessment to determine the organizational respect for nuclear safety, and (3) identifying and resolving abnormal plant conditions or indications that cannot be readily explained. INPO shared SOER 02-4 with WANO, which republished it as a WANO document.

Safety culture is thoroughly examined during each plant evaluation. INPO expects each evaluation team to evaluate safety culture throughout the process, including during the preevaluation analysis of plant data and observations made at the plant. The results of this review are included in the summary on organizational effectiveness and may be documented as an area for improvement, as appropriate. The INPO evaluation team discusses aspects of a plant's safety culture with the CEO of the utility at each evaluation exit briefing.

7. Cornerstone Activities

a. Evaluation Programs

Members host regular INPO evaluations of their nuclear plants approximately every 2 years. The INPO evaluation teams periodically conduct additional evaluative review visits on corporate support and on other more specific areas of plant operation. During these evaluations and reviews, the INPO teams use standards of excellence based on the POCs, their own experience, and their broad knowledge of industry best practices.

This approach shares beneficial industry experience while promoting excellence in the operation, maintenance, and support of operating nuclear plants. Written POCs, developed by INPO with industry input and review, guide the evaluation process and are the bases for identified areas for improvement. The evaluations are performance oriented and emphasize the results achieved and the behaviors and organizational factors important to future performance. The evaluations focus on those issues that affect nuclear safety and plant reliability.

i. Plant Evaluations

Teams of approximately 18 to 25 qualified and experienced individuals conduct evaluations of operating nuclear plants that focus on plant safety and reliability. In 2012, U.S. utilities received 39 plant evaluations or WANO peer reviews. The evaluation teams include senior reactor operators, other peer evaluators from different utilities, host utility peer evaluators, and an executive industry advisor. The scope of the evaluation includes the following functional areas:

- operations
- maintenance
- engineering
- radiological protection
- chemistry
- training

In addition, the teams evaluate cross-functional performance areas (i.e., processes and behaviors that cross organizational boundaries) and address process integration and interfaces. The teams evaluate the following cross-functional areas:

- safety culture
- operational focus
- configuration management
- equipment reliability and work management
- performance improvement (learning organization)
- organizational effectiveness

As part of the evaluation process, an evaluation team evaluates important aspects of a site's quality assurance programs to ensure that these programs provide confidence that the plant is satisfying the requirements for activities important to nuclear safety.

Team leaders lead and coordinate team activities and provide a focal point for the evaluation of station management and leadership by concentrating on evaluating leadership, organizational effectiveness, safety culture, and nuclear oversight topics.

A key part of each evaluation includes the performance of operations and training personnel during simulator exercises. In addition, the evaluation also includes, where practicable, observations of refueling outages, plant startups, shutdowns, and major planned evolutions.

The evaluation team provides the utility with formal reports of strengths and areas for improvement and a numerical rating of overall plant performance. As part of the 1983 annual INPO CEO workshop, INPO prepared a set of indicators for each nuclear station that reflected station participation in and commitment to INPO programs. INPO provided this information to each CEO. One of these indicators was an assessment of each station's overall performance based on INPO evaluations and on the judgment of INPO team managers and senior management.

With the approval of the Board of Directors, INPO decided that it would assess the overall station performance in the context described above after each evaluation and that it would share this assessment privately with the CEO at the exit meeting. Eventually, the institute developed a numerical assessment and now provides each station an assessment from Category 1 (excellent) to Category 5, which is defined as the level of performance at which the margin to nuclear safety is substantially reduced. Such a process reflects the desire of utility managers to know more precisely how their stations' performance compares relative to the standards of excellence. In addition, this process is in accordance with INPO's responsibility to the individual CEO and to its members for identifying low-performing nuclear plants and for stimulating improvement in performance.

Even though standards for performance have risen substantially over the years, the number of plants in Categories 1 and 2 has remained relatively constant, even as standards of excellence have improved. Additionally, several conclusions can be drawn from evaluations over the years. Excellent plants (Category 1) and Category 2 plants show strong leadership, are self-critical, do not tolerate complacency, are operationally focused, have exceptional equipment performance, and effectively use training to improve performance. Attributes of Category 3 and 4 stations may include leaders who do not set high standards, a weak self-critical attitude, weak day-to-day operations, broad equipment problems, and deficient fundamental knowledge and skills in several areas. INPO has not assessed a station as Category 5 in over a decade.

The final report includes utility responses to the identified areas for improvement and their commitments to specific corrective action. In subsequent evaluations and other interactions, INPO specifically reviews the effectiveness of actions taken to implement these improvements.

In addition to the strengths and areas for improvement provided in the evaluation report, subjective team comments are often communicated to the member CEO during the evaluation exit meeting. The intent of these comments, which are often more intuitive, is to help the utility recognize and address potential issues before they adversely affect actual performance. Copies of the plant's evaluation report are distributed according to a policy approved by the institute's Board of Directors.

The industry also hosts WANO peer reviews conducted by the WANO-Atlanta Centre. These peer reviews are conducted at each U.S. station approximately every 6 years and are performed in place of an INPO plant evaluation at each station. These peer reviews use a methodology similar to that of plant evaluations, but with teams that include international peers.

Numerous improvements have been made in plant safety and reliability as a result of addressing issues identified during evaluations, peer reviews, plant self-assessments and comparison and emulation among plants. The time that plants operate versus the amount of time that they are shutdown has improved significantly, the frequency of unplanned shutdowns has decreased markedly, and the reliability and availability of safety systems has improved measurably.

ii. Corporate Evaluations

Member utilities that operate multiple nuclear stations request that INPO conduct corporate evaluations on an interval of 4 to 6 years. INPO conducts corporate evaluations at single nuclear station utilities when such evaluations are requested by the utility or when they are deemed necessary by the institute. The INPO-conducted corporate evaluations reflect the important role of the company headquarters in supporting the successful operation of plants within a multisite fleet. INPO conducted four corporate evaluations in 2012.

A tailored set of POCs defines the scope of activities and the standards for corporate evaluations. The corporate evaluation focuses on the impact that the corporation has on the safe operation of its nuclear plants. Areas typically evaluated during a corporate evaluation include the following:

- direction and standards for station operation, including the organizational alignment, communications, and accountability for strategic direction, business and operational plans, and performance standards
- governance, monitoring, and independent oversight of the nuclear enterprise
- support for emergent station issues and specialty areas such as major plant modifications, including replacement of steam generator and reactor vessel heads and station upgrades to extract more power and efficiency
- performance of corporate functions, such as human resources, industrial relations, fuel management, supply chain management, and other areas applicable to the nuclear organization

INPO members use corporate evaluation results to help ensure that essential corporate functions are providing the leadership and support necessary to achieve and sustain excellent nuclear station performance. As a consequence of responding to issues identified during corporate evaluations, stations often have refocused appropriate resources and leadership attention on improving station safety and reliability.

At the request of its members, INPO meets with utility boards of directors to provide an overview of plant, and fleet performance, when applicable. The boards of directors use these briefings as an input to their assessment of operational risk.

iii. Other Review Visits

The industry also uses INPO to conduct review visits in selected industrywide problem areas to supplement the evaluation process. These visits are typically initiated by INPO and are evaluative in nature. The results of review visits may be used as an input to the evaluation process. The visits are designed as indepth reviews of technical areas that could have a significant impact on nuclear safety and reliability. Such areas include critical materials issues that affect the structural integrity of the reactor coolant system and reactor vessel internals of both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Other areas include components or systems that are significant contributors to unplanned plant transients and forced loss rate, including main generator and transformer, switchyard, and electrical grid components. In 2012, INPO conducted 97 review visits.

Similar to plant evaluations and peer reviews, review visits evaluate station performance against the INPO POCs to a standard of excellence. In some areas, such as materials, industry groups have developed detailed technical guidance that each utility has committed to implement. The materials review visit teams also use this guidance to ensure that program implementation is consistent and complete and meets the industry-developed standards.

Review visit teams are led by an INPO employee and include industry personnel who have unique expertise in the area of the review that is not typically within the skill set of INPO members of plant evaluation or peer review teams. Review visits typically include a week of preparation followed by a week on site.

Review visit reports contain beneficial practices and recommendations for improvement. These reports are sent to the station site vice president. For potential safety-significant recommendations, INPO may request a response. The subsequent plant evaluation or WANO peer review team follows up on each of the recommendations that require a response to ensure that identified issues are addressed. Periodically, INPO compiles the beneficial practices and recommendations and posts the information on the secure member Web site to allow all utilities to benchmark their programs.

The following sections discuss the details of selected review visit programs.

Pressurized-Water Reactor Steam Generator Review Visits

INPO initiated steam generator review visits in 1996. In the early 1980s, steam generator tube leaks and ruptures were significant contributors to lost power generation and were the cause of several events deemed significant by INPO. The industry as a whole became more sensitive to the importance of steam generator integrity as a contributor to core damage frequency analysis. The industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program, developed and maintained detailed guidance on qualification and implementation of nondestructive testing techniques, engineering assessments of steam generator integrity, and detection and response to tube leakage and ruptures. In mid-1995, the industry requested that INPO help improve the prevention and detection of steam generator degradation by verifying correct and consistent implementation of industry guidance at individual stations and by evaluating steam generator management programs against standards of excellence. As a result, INPO established the Steam Generator Review Visit Program. Other review visits that were initiated later used the steam generator review visit process as a model.

Steam generator review visits focus on steam generator inservice inspection and repair; use of qualified personnel and techniques for eddy-current examinations of tubes; tube plugging procedures; assessment of current inspection results; chemistry conditions that affect steam generators; and steam generator primary-to-secondary leak detection, monitoring, and response.

In general, steam generator management programs have steadily improved. In addition, stations have been implementing these programs effectively as evidenced by the lack of safety-significant events and events that contribute to lost generation. Steam generator replacements have also contributed to overall improved performance. Consequently, steam generator review visits currently identify few significant issues. However, the review visits have identified a need for improved timeliness in implementing industry-developed or revised guidance and for improved rigor in inspecting for loose part and in evaluating and retrieving them.

Boiling-Water Reactor Vessel and Internals Review Visits

In 2001, INPO initiated BWR vessel and internals review visits at the request of the industry. In the early 1990s, vessel and internal issues caused by intergranular stress-corrosion cracking became significant contributors to lost power generation. Safety concerns associated with this degradation prompted the industry to form the EPRI BWR Vessel and Internals Project. This group developed detailed guidance to address inspection, mitigation, repair, and evaluation of degradation for components important to safety and reliability.

BWR vessel and internals review visits focus on nondestructive examinations; inspection scope and coverage; evaluation of crack growth and critical flaw size; effectiveness of strategies to mitigate intergranular stress-corrosion cracking, including hydrogen addition and application of noble metals; and chemistry conditions that affect long-term health, including potential effects on fuel.

Industry overall performance has improved as evidenced by the lack of safety-significant events and events that contribute to lost generation.

Pressurized-Water Reactor Primary Systems Integrity Review Visits

INPO initiated PWR primary systems integrity review visits in 2003. Since the early 1980s, a number of notable events associated with leakage from PWR borated systems have resulted in additional oversight by the NRC and INPO. In some cases, these leakage events have resulted in corrosion and wastage of pressure-retaining components in the reactor coolant system. The EPRI PWR Materials Reliability Program was formed as an industry initiative in 1998 to develop guidance to address materials degradation issues. Because of the importance of primary systems integrity, INPO began performing indepth review visits focused on boric acid corrosion control and Alloy 600 degradation management, including dissimilar metal butt welds.

PWR primary systems integrity review visits focus on the inspection and evaluation of pressure-retaining components in the reactor coolant system; the qualification of nondestructive examination personnel and techniques; and the monitoring of and response to unidentified leakage in containment, including management guidance and operator procedures.

As a result of these industry efforts, performance appears to be improving. Stations are identifying degradation before leakage occurs. Stations have also more aggressively pursued indications of minor unidentified leakage. Alloy 600 dissimilar metal butt weld examinations and mitigation will continue over the next few years as the enhanced industry-defined actions continue to be performed and as inspections take full advantage of improved nondestructive examination techniques.

Transformer, Switchyard, and Grid Review Visits

INPO initiated transformer, switchyard, and grid review visits in 2004. Many transformers have been in service for numerous years and are often the original station transformers. Considering this aging—along with the recent trends in power uprates, license renewal, and increased loading—these transformers may be operating with a reduction in margin. With this decrease in margin, the need for increased monitoring, trending, and predictive and preventive maintenance became apparent in order to identify and mitigate potential problems before they result in online failure. Additionally, a series of events in 2003, including the blackout in northeastern United States and parts of Canada, reinforced the need for nuclear plants to have reliable offsite power. In addition, renewed focus on how nuclear plant conditions and electrical power system lineups to the switchyards can help minimize and prevent grid events.

The transformer, switchyard, and grid review visits focus on communication and coordination with grid operators, including formal agreements and implementing procedures, adequacy of offsite power, and predictive and preventive maintenance for large power transformers and switchyard equipment.

Although isolated events related to switchyards, transformers, and grids continue to occur, additional rigor in maintenance and interfaces has shown some improvement. Additionally, sharing of information and lessons learned among utilities is resulting in implementation of barriers to prevent future events. The continuation of review visits should reduce the number and significance of events.

Main Generator Review Visits

The industry initiated main generator review visits in 2004 after the identification of an adverse trend involving failures of main generators and related support systems. The number of main generator failures that hindered power production or extended an outage, or both, had doubled from 1999 to 2003. During this time, unplanned scrams caused by generator problems increased to around five per year from the previous average of two per year. The most frequent generator maintenance challenges involved support systems, such as stator cooling water and the exciter, and often included human performance elements. As a result of industry identification of this adverse performance, INPO began conducting main generator review visits to focus on improving the performance of main generators.

Main generator review visits focus on performance and condition monitoring to ensure that the generator is operating within design parameters and to detect early signs of equipment degradation, preventive and condition-based maintenance to address the effects of aging, outage planning to ensure that important main generator work is performed, and knowledge and skill levels of personnel to ensure proper workmanship.

Emergency Preparedness

In 2007, INPO reestablished its emergency preparedness section to help the industry continue to improve its readiness to respond to radiological and other site emergencies. INPO began this initiative in response to a need identified in 2002 by the Nuclear Energy Institute (NEI) and a subsequent industry review led by INPO of 25 plants over 3 years. These visits identified opportunities for improvement that included more timely and accurate classifications, notifications, and protective action recommendations; strengthened drill programs; and increases in emergency response organization staffing. The Emergency Preparedness Review Visit Program is a formal INPO program in which each site receives a visit every 4 years.

In 2012, INPO entered its sixth year of conducting emergency preparedness review visits. During this time, the institute identified several industrywide issues that INPO-facilitated working groups comprising industry leaders are addressing. INPO developed and published a guideline that provides a basic task analysis and training program elements for key emergency response organization members. The institute is drafting additional guidance on how to better control equipment important to emergency preparedness and how to develop realistic training and evaluation of shift manager oversight during emergencies. INPO anticipates that published guidance on these topics will be available to the industry in 2010.

The INPO emergency plan and the recently updated Emergency Response Center is used to assist members in mobilizing the resources of the nuclear industry and to provide other resources or assistance as necessary, following the classification of an emergency event. In 2013, INPO will complete an emergency response drill it will perform with support from an industry fleet emergency preparedness organization. This drill will demonstrate the value of a collaborative relationship with industry members in providing needed support.

b. Training and Accreditation Programs

The U.S. commercial nuclear power industry strongly believes that proper training of plant operators, maintenance workers, and other support group workers is of paramount importance to the safe operation of nuclear plants. As a result, the industry established the Academy in 1985 to operate under the responsibility of INPO. The industry formed the Academy to focus and unify high standards in training and qualification and to promote professionalism of nuclear plant personnel. The Academy integrates the training-related activities of all members, the independent National Nuclear Accrediting Board, and the institute. Through INPO, the Academy conducts seminars and courses and provides other training and training materials for utility personnel.

All U.S. nuclear plants have accredited training programs and are branches of the Academy. A utility becomes a member of the Academy when all of its operating plants have achieved accreditation for all applicable training programs.

INPO interacts with all members in preparing for, achieving, and maintaining accreditation of training programs for personnel involved in the operation, maintenance, and technical support of nuclear plants. These interactions are similar in content to the accreditation efforts of schools and universities and include evaluations of accredited training programs, activities to verify that the standards for accreditation are maintained, and assistance at the request of member utilities. Written objectives and criteria are jointly developed with the industry and guide the accreditation process.

Unlike its role in the plant evaluation and assessment process described above, INPO is not the accrediting agency. The independent National Nuclear Accrediting Board examines the quality of utility training programs and makes all decisions on accreditation. If training programs meet accreditation standards, the National Nuclear Accrediting Board awards or renews accreditation. If significant problems are identified, it may defer initial accreditation, place accredited programs on probation, or withdraw accreditation. Accreditation is maintained on an ongoing basis and is formally renewed for each of the training programs every 4 years. The National Nuclear Accrediting Board comprises training, education, and industry experts. It is convened and supported by INPO; however, it is independent in its decisionmaking authority. National Nuclear Accrediting Board members are selected from a pool of individuals from utilities, postsecondary education, nonnuclear industrial training, and NRC nominations. Each National Nuclear Accrediting Board consists of five sitting members, with a maximum of two utility representatives to ensure its independence from the nuclear industry.

The accreditation process is designed to identify strengths and weaknesses in training programs and to assist in making needed improvements. The process includes selfevaluations by members with assistance provided by the INPO staff, onsite evaluations by teams of INPO and industry personnel, and decisions by the independent National Nuclear Accrediting Board. Members should seek and maintain accreditation of training programs for the following positions or skill areas:

- shift managers
- senior reactor operators
- reactor operators
- nonlicensed operators
- continuing training for licensed personnel
- shift technical advisors
- instrument and control technicians and supervisors
- electrical maintenance personnel and supervisors
- mechanical maintenance personnel and supervisors
- chemistry technicians
- radiological protection technicians
- engineering support personnel

In 2002, the industry updated the accreditation objectives to place additional emphasis on training for performance improvement. In striving for excellence, the industry recognized that training must be an integral part of each plant’s business strategy and daily operations to ensure a highly skilled workforce. This approach strengthens the link between the analysis of performance gaps and the training that results in tangible improvements in people and plant activities. The five-step systematic approach to training remains the essential tool for providing training that is results oriented. Both line and training organizations are expected to work together to analyze performance gaps and to design, develop, and deliver training that enhances knowledge and skills to measurably improve plant performance. Such an approach to improving worker knowledge and skills contributes to high levels of safety as seen in industry gains in equipment reliability, safety system availability, collective radiation exposure, worker safety, and fewer events. The role of training will continue to be vital in the coming years as many experienced workers retire and as new workers enter the workforce.

Although the accreditation process is independent of the NRC, the agency recognizes and endorses the process as a means for satisfying regulatory training requirements. In its annual report entitled, "Annual Report on the Effectiveness of Training in the Nuclear Industry," the NRC noted that "monitoring the INPO managed accreditation process continued to provide confidence that accreditation is an acceptable means of ensuring the training requirements contained in 10 CFR [Part] 50 and 10 CFR [Part] 55 are being met." In addition, the NRC assessment of the accreditation process indicates that continued accreditation remains a reliable indicator of a successful systematic approach to training implementation and contributes to the assurance of public health and safety by ensuring that nuclear power plant workers are being trained appropriately.

i. Training and Qualification Guidelines

The Academy develops and distributes training and qualification guidelines for operations, maintenance, and technical personnel. These guidelines are designed to assist the utility in developing quality training programs and in selecting key personnel.

Training and qualification guidelines are revised and updated periodically to incorporate changes to address industry needs and to take into account lessons learned from other INPO programs such as evaluations, events analyses, working meetings, and workshops. These training and qualification guidelines provide a sound basis for utility training programs.

ii. Courses and Seminars

The industry benefits extensively from courses and seminars that the Academy conducts to help personnel better manage nuclear technology, more effectively address leadership challenges, and improve their personal performance. In 2012, nearly 1,400 industry employees, including many international representatives, participated in more than 70 courses and seminars. Examples of courses and seminars conducted are as follows:

- Goizueta Director's Institute (focused on the directors of member boards) (INPO, in partnership with the Goizueta Business School of Emory University, conducts the course entitled, "The Impact of Governance on the Nuclear Power Industry," a nuclear education course designed for directors in the nuclear industry
- reactor technology course for utility executives
- senior nuclear executive seminar
- senior nuclear plant management course

- human performance fundamentals course
- high performance teamwork development
- operations supervisor professional development seminar
- first-line leadership seminar
- next-level leadership seminar
- seminars for new plant managers and for new managers in operations, radiological protection, chemistry, maintenance, engineering, nuclear oversight, and training

In February 2006, INPO launched the National Academy for Nuclear Training e-Learning (NANTeL) system. Using Web-based technologies that allow distance learning, the NANTeL system training includes courses and proctored examinations for plant access, radiation worker, human performance, and industrial safety qualification to industry standards. By July 2006, all member utilities had agreed to participate in the NANTeL system by accepting generic training and updating the industry's Personnel Access Data System for completion of the training courses. The NANTeL system offers 42 generic and 215 utility or site-specific training courses. Between March 1, 2006, and December 31, 2012 more than 110,000 industry workers have completed more than 1 million courses.

Meeting the challenges of developing a well-trained, knowledgeable workforce in the future continues to receive attention. In early 2008, INPO began work on the first phase of a new industry initiative called the Future of Learning. Developed with extensive industry participation, this initiative lays out a strategy to guide training efforts in the years ahead. It will help the industry deal with workforce renewal, the training of a new generation of workers, and the training of even more workers to support new plant construction.

INPO efforts to help prepare and energize the nuclear workforce of tomorrow include a new leadership seminar designed for emerging nuclear leaders. In addition, the course entitled, "Nuclear Citizenship for New Workers," that emphasizes the uniqueness of the Nation's nuclear industry and an industrywide instructor training and certification program that uses a blend of distance learning and classroom instruction are now available.

c. Analysis and Information Exchange Programs

The analysis and information exchange programs improve plant safety by identifying the causes of industry events that may be precursors to more serious events. Stations are required to share operating experiences and lessons learned with INPO. INPO then analyzes and rapidly communicates the information to the industry through a variety of methods and products. In addition, INPO analyzes a variety of operational data to detect trends in industry performance and communicates the results to the industry.

INPO operates and maintains extensive computer databases to provide members and participants ready access to information on plant and equipment performance and operating experience. These databases are accessible from INPO's secure member Web site. For example, the industry uses Nuclear Network®, a worldwide Internet-based communication system, to exchange information on the safe operation of nuclear plants.

WANO also uses Nuclear Network® as a primary means for communicating and exchanging operating experience among its members and regional centers.

i. Events Analysis Program

INPO reviews and analyzes operating events from both domestic and international nuclear plants through its Significant Event Evaluation and Information Network (SEE-IN) Program. The program is designed to provide indepth analysis of nuclear operating experience and to apply the lessons learned across the industry. Events are screened, coded, and analyzed for significance; those with generic applicability are disseminated to the industry in one or more of the following forms, beginning with events of greatest importance:

- SOERs
- significant event reports (SERs)
- significant event notifications (SENs)

Members support the events analysis program by providing INPO with detailed and timely operating experience information. Operating experience information is freely shared among INPO members. The U.S. industry submits more than 2,000 operating experience entries every year, or about 30 to 40 per station. These entries enable a single station to multiply its experience base for identifying problems. This experience base includes safety systems, which have similar components across many stations. For example, one station recently discovered the scoring of a cylinder on an emergency diesel generator (EDG) that could have rendered it inoperable. Other stations were able to use this information to take actions to inspect their EDGs before actual equipment malfunction. A key to this success is the timeliness of reporting. Stations typically report events in less than 50 days after occurrence.

Members are required to evaluate and take appropriate action on recommendations provided in SOERs. During onsite plant evaluations, INPO teams follow up on the effectiveness of each station's actions in response to SOER recommendations. For example, during a recent plant evaluation, team members reviewing SOER recommendations identified a potentially significant transformer problem that likely would lead to catastrophic failure if it was not corrected in a timely manner. This event was avoided because of lessons documented in an SOER. Topics of SOERs in recent years include loss of grid, reactivity management, reactor core designs, transformers, unplanned radiation exposures, and rigging and lifting of heavy loads.

Members should review and take actions, as appropriate, on SENs, SERs, and other reports provided by INPO. INPO evaluates the effectiveness of utility programs in extracting and applying lessons learned from industrywide, and internal station operating experience.

INPO maintains all operating experience reports since the start of the SEE-IN program in searchable databases available on the secure member Web site. This information supports members in applying historical lessons learned as new issues are analyzed or activities are planned. INPO also provides "just-in-time" briefing summaries in numerous topical areas in a format designed to help plant personnel prepare to perform specific tasks. These documents provide ready-to-use materials to brief workers on problems experienced and lessons learned during recurring activities.

ii. Other Analysis Activities

INPO analyzes industry operational data from a variety of sources—events, equipment failures, performance indicators, and regulatory reports—to detect trends in industry performance. INPO communicates the results of analyses to the industry using several methods, including topical reports. These documents typically review events and other data over a period of years to summarize performance trends and causes and suggest actions. Subjects of recent topical reports include fuel reliability, foreign material intrusion, intake cooling blockage, large motor failures, and contractor personnel performance. Stations use these reports to assess their performance and to identify improvements. In addition, individual plant performance data are analyzed, and the results are used to support other INPO activities, such as evaluations and assistance.

iii. Nuclear Network® System

Nuclear Network is an international electronic information exchange for sharing nuclear plant information. It is the major communication link for the SEE-IN and WANO event reporting system. The system transmits operating experience information, SERs, and other nuclear technical information.

The system includes a special dedicated method for reporting unusual plant situations. This feature allows the affected utility to provide timely information simultaneously to all Nuclear Network® users, including the U.S. industry, INPO's international and supplier participants, and WANO members, so the affected station does not have to respond to multiple inquiries. In addition, members are promptly informed of problems occurring at one station, allowing them to implement actions to prevent a similar occurrence.

iv. Performance Data Collection and Trending

INPO operates and maintains a consolidated data entry system as a single process for the collection of data and information related to nuclear plant performance. Members provide routine operational data in accordance with the WANO Performance Indicator Program or regulatory requirements on a quarterly basis. These plant data are then consolidated for trending and analysis purposes. Industrywide data, and trends developed from the data are provided to member and participant utilities for a number of key operating plant performance indicators. Members use these data for comparison and emulation with other plants, in setting specific performance goals, and in monitoring and assessing the performance of their nuclear plants.

In the mid-1980s, the industry worked with INPO to establish a set of overall performance indicators focused on plant safety and reliability. These indicators have gained strong acceptance and use by utilities to compare performance, set targets, and drive improvements. Examples of indicators collected and trended include unplanned automatic scrams, safety systems performance, unit capability factors, forced losses of generation, fuel reliability, collective radiation exposure, and industrial safety accidents.

The industry has established long-term goals for each indicator on a 5-year interval, beginning in 1990. Annex 2 of this report provides key performance indicator graphs for U.S. plants.

v. Equipment Performance Data

INPO operates and maintains the EPIX system, which tracks the performance of equipment important to safety and reliability. The industry reports equipment performance information to EPIX in accordance with established guidance. Member utilities use the data to identify and solve performance problems of plant equipment with the goal of enhancing plant safety and reliability. INPO also uses the information for performance trending to identify industrywide performance problems. the institute also makes the data available to the NRC to support equipment performance reviews by the regulator.

vi. Operating Experience for New Plant Construction

In 2009, a means for collecting and distributing experience from construction problems was established through the U.S. industry's Nuclear Network® system. Nuclear Network® has long been the forum for rapid and secure communications and has hosted the industry's operating experience program. The New Plant Construction Program has a similar mission to that of the operating experience; however, it is tailored to the unique needs of utilities with construction projects. The New Plant Construction Program has since been upgraded to include work at Watts Bar Nuclear Plant, Unit 2, and the new construction units at the Vogtle Electric Generating Plant and Virgil C. Summer Nuclear Station.

d. Assistance Programs

Between evaluations, a station can request and receive assistance in specific problem areas to help improve plant performance. In addition, INPO monitors the performance of member utility stations between evaluations to identify areas in which assistance can be used to improve plant performance or to respond to declining performance. The purpose of this monitoring is to identify, as early as possible, stations that exhibit indications of declining performance so that focused assistance can be provided to help reverse the performance trend. INPO also provides members with comparisons of their plants' performance to overall industry performance in a variety of areas.

A majority of assistance visits to member utilities by INPO personnel and industry peers are at the request of the stations. This assistance is targeted for specific technical concerns and for broader management and organizational issues. Although assistance is generally requested by a station, INPO may, in some cases, suggest assistance in a specific area to stimulate improvements.

Assistance resources are provided using a graded approach that provides a higher priority to those plants that need greater performance improvement. An INPO management senior representative is assigned to each station to facilitate assistance efforts. Station and utility management maintains close liaison with the senior representative to help identify areas for which INPO resources can best be used to address specific issues and to help improve overall station performance.

When significant performance shortfalls persist at a station or when performance trends indicate chronic conditions that could detract from safe and reliable plant operation, INPO will follow a policy of graduated engagement with the member utility. For a nuclear plant that shows either consistently poor performance over several evaluation cycles or a significant decline in performance between evaluation cycles, the INPO staff will recommend and obtain concurrence from the INPO CEO to include the plant in a special focus category. For plants that need special focus, INPO will establish a Special Focus Oversight Board that will conduct scheduled periodic reviews to determine the effectiveness of station improvement activities and to provide rapid feedback. Board members will usually include both industry and INPO executives.

INPO provides documents that describe nuclear safety principles, effective leadership and management practices, and good work processes and practices to assist member utilities. Members help INPO develop these documents and then use them to address specific improvement needs.

The conduct of workshops, seminars, working meetings, and other activities is also done to assist in the exchange of information among members and to support the development of industry leaders and managers.

INPO facilitates information exchange among member utilities by identifying and cataloging information on a wide range of activities that stations are doing especially well. The information on effective programs and practices is shared with members on request and through a number of other forums. This assistance fosters comparison and the exchange and emulation of successful methods among members.

i. Assistance Visits

Members may request assistance visits in specific areas of nuclear operations in which INPO personnel have experience or expertise. INPO personnel and industry peers normally conduct such visits. For example, if a member requests assistance in some specific aspect of maintenance, INPO will include a peer from another plant that handles that aspect of maintenance particularly well. INPO provides written reports that detail the results of the visits to the requesting utility. In most cases, the assistance visit includes actual methods and plans for improving performance as part of the assistance visit.

In 2012, INPO provided more than 150 assistance visits using over 100 industry peers. Key areas of assistance provided included operational focus, maintenance and work management, engineering programs, chemistry, radiological protection, human performance, and industrial safety. Additional areas of assistance conducted in 2012 involved supplier participants, with a focus on supplemental personnel and fuel performance. In addition to assistance visits to stations for specific functional areas during 2012, senior representatives conducted over 140 visits to their assigned stations to interact with station management and to monitor for early signs of performance decline. INPO teams led by senior representatives made multiple assistance visits at stations designated as special focus.

Effectiveness reviews performed by INPO approximately 6 months after assistance visits show that assistance visits are highly valued by station management and are contributing to improved performance.

ii. Development of Documents and Products

Several categories of documents and other products are designed and developed to help member utilities and participants achieve excellence in the operation, maintenance, training, and support of nuclear plants. INPO documents and products include the following key categories:

- The POCs, revised in 2012 in collaboration with WANO, are standards for plant and corporate performance used to promote excellence in the operation, maintenance, and support of operating nuclear electric generating stations. the POC document is the standard used in INPO evaluation activities, and member utilities often use it in self-evaluations.

  The POCs support the achievement of the following set of operational excellence outcomes:

  - sustainable, high-level plant performance
  - sustainable, event-free operation
  - avoidance of unplanned, long-duration shutdowns
  - well-managed and understood safety, design, and operational margins
  - high levels of plant worker safety
  - a highly skilled, knowledgeable, and collaborative workforce

- Principles documents address professionalism, management and leadership development, human performance, and other cross-functional topics important in achieving sustained operational excellence. INPO prepares these documents with substantial involvement of industry executives and managers. The principles extracted from the documents are used extensively in evaluation and assistance activities.

  The first of the principles documents entitled, “Principles for Enhancing Professionalism of Nuclear Personnel,” addresses human resource management areas focused on developing nuclear professionals and includes personnel selection, training and qualification, and career development. Two supplemental documents— “Management and Leadership Development,” and “Excellence in Human Performance” —build on the original document. Utility executives use the document entitled “Management and Leadership Development,” to assist in the identification, development, assessment, and selection of future senior managers. The document entitled “Excellence in Human Performance,” provides practical suggestions for enhancements in the workplace that promote excellent human performance.

  In 1999, INPO distributed “Principles for Effective Self-Assessment and Corrective Action Programs,” which emphasizes the importance of establishing a self-critical station culture and identifying the key elements of effective self-assessment and corrective action programs.

- Guideline documents establish the bases for sound programs in selected areas of plant operation, maintenance, training, and cross-functional areas of direct importance to the operation and support of nuclear stations. Guidelines assist members in meeting the objectives used in evaluations and accreditation. The guidelines are recommendations based on generally accepted industry methods. They are not directives; instead, the intent of these guidelines is to help utilities maintain high standards. Although member utilities do not have to follow each specific method described, they are expected to strive to meet the intent of INPO guidelines.

- INPO provides good practices, work process descriptions, nuclear exchange documents, and other documents to assist members. Typically, these documents are developed from programs of member utilities and INPO's collective experience. INPO synthesizes the information into a document by its staff, with industry input and review. In general, the documents define one method of meeting INPO POCs in specific areas, although other programs or methods may be as good or better. Utilities are encouraged to use these documents in developing or improving programs applicable to their plants. These documents can be used in whole or in part, as furnished, or modified to meet the specific needs of the plant involved.

INPO produces various other documents, such as analysis reports and special studies, as needed. Other assistance products include lesson plan materials, computer-based and interactive video materials, videotapes, and examination banks. The Academy' quarterly magazine, *The Nuclear Professional,* features how plant workers have solved problems and have made improvements that enhanced safety.

iii. Workshops and Meetings

INPO sponsors workshops and working meetings for specific groups of managers on specific technical issues as forums for information exchange. This exchange provides an opportunity for INPO and industry personnel to discuss challenges, performance issues, and areas of interest. It also allows individuals from INPO members and participants to meet and exchange information with their counterparts. In 2012, nearly 1,100 industry personnel participated in more than 70 meetings and workshops.

8. Key Initiatives 2010 – 2014

The nuclear industry continues to change and move at a demanding pace. New technologies, new people, and plans for new plants are adding even more challenges to the mix. The future will bring with it new demands for INPO and its members.

Cross-functional INPO teams began developing a strategic plan in mid-2008, building on the success and lessons learned from the previous plan. The development of this plan was done by taking into account the needs of stakeholders and by focusing on key areas in which INPO wants to have significant impact in the coming years. The strategic plan is centered on the following ten outcomes that are top priority:

(1) Fundamentals. Fundamentals are ingrained and rigorously applied.
(2) Operational Risk. Industry workers effectively measure, mitigate, and manage operational risk.
(3) POC Revision. INPO evaluations of plant and corporate performance reflect global standards of excellence.
(4) International Safety and Reliability. Global nuclear safety and reliability are improved.
(5) Improvement of Industry Knowledge and Skills. Industry worker performance is improved through quality training.
(6) Learning from Fukushima. Plant vulnerabilities are reduced. A report published by the NRC (NUREG-1650, "United States National Report for the Convention on Nuclear Safety," Revision 4, "The United States of America National Report for the 2012 Convention on Nuclear Safety Extraordinary Meeting," issued July 2012) documents details about INPO's plan on addressing lessons learned from Fukushima.
(7) Early Detection and Response To Declining Performance. Declines in station performance are detected early and corrected quickly.
(8) Knowledge and Skills of INPO Employees. INPO's competent, engaged workforce adapts effectively to changing demands.
(9) INPO Governance. INPO effectiveness is improved through defined governance and oversight.
(10) Knowledge Management. INPO uses information and knowledge effectively to identify early industry trends and emerging issues.

The 5-year business plan is built around high-priority organizational themes that are critical for accomplishing INPO's vision. These themes are cross-functional and transcend cornerstone, division, and department boundaries. The plan is not a checklist of activities or projects that INPO does; instead, it is a plan that describes the outcomes INPO intends to produce or influence.

The industry continuously provides feedback to INPO on issues that affect station operation. Many INPO initiatives are based on industry trends and important focus areas. One initiative that is underway is described below.

a. New Plant Design and Construction

For many years, no new nuclear plants have been built in the United States. However, because of the need for additional power, concerns over the environmental effects of carbon-based fuels, the streamlined licensing process, and financial incentives provided by the Energy Policy Act of 2005, U.S. utilities are once again planning new plant construction. To support this effort, INPO formed a new plant deployment group in 2006 to engage with the nuclear industry and to plan for institute's involvement though application of its cornerstone programs.

In 2006, INPO updated a report entitled, "Operating Experience to Apply to Advanced Light Water Reactors," which includes lessons learned from significant events. The updated report includes experience from operations and maintenance activities that the design of new plants should address. INPO participant plant designers and utility groups are using this document in their review of the new designs.

INPO also engaged utilities planning to submit license applications in a series of benchmarking trips in 2006 and 2007 to international utilities and plant designers in France and Japan, an aircraft company, and a coal plant with advanced control systems. These trips provided an opportunity to learn more about new technologies that have evolved since the last period of nuclear plant construction, most notably in plant standardization, computerized man-machine interface, and modular construction. INPO is issuing a report to its members that features the information gathered from these trips.

In an effort to support the human factors of new reactor designs, INPO has worked with its members and several architect engineers to evaluate the proposed man-machine interface for the new technology and designs of Generation 3 reactors. Workshops have been helpful in providing input to the design engineers on how to optimize this interface and how to address new techniques to ensure strong human performance when using the new designs.

INPO is currently not involved in discussions on the siting of new plants.

9. Relationship with World Association of Nuclear Operators

U.S. nuclear utilities are represented in WANO through INPO. As such, INPO coordinates the U.S. nuclear utilities' activities in WANO. INPO also provides operational support and facilities for the WANO-Atlanta Centre, one of the four WANO global regional centers. The WANO-Atlanta Centre Governing Board usually appoints an INPO executive to serve as the Atlanta Centre director.

WANO-Atlanta Centre contracts with INPO to provide resources in terms of seconded staff to support the Centre's day-to-day operations. WANO-Atlanta Centre also contracts with INPO to provide administrative support services, such as payroll, computer support, and employee benefit administration.

WANO-Atlanta Centre activities and programs include the following:

- WANO teams of U.S. and international peer reviewers conduct reviews at the request of INO members to identify strengths and areas for improvement associated with nuclear safety and reliability. A WANO peer review conducted at a U.S.-INPO member plant is performed in place of an INPO plant evaluation.
- The WANO exchange of operating experience information provides detailed descriptions of events and lessons learned to member utilities worldwide.
- WANO collects, trends, and disseminates performance indicator data to facilitate goal setting and performance trending and to encourage emulation of the best industry performance.
- WANO conducts technical support missions to allow direct sharing of plant operating experience and ideas for improvement.

- WANO designs professional and technical development courses, seminars, and workshops to enhance staff development and to share operating experience.

The U.S. industry and INPO receive a substantial benefit through their relationship with WANO and the international nuclear community. Many improvements have been implemented in the U.S. based on lessons learned from the more than 340 units that exist outside of the United States. INPO works to remain fully aware of trends in the global nuclear industry and continues to strengthen relationships in this area.

10. Industry Response to the Accident at Fukushima

The earthquake and tsunami in Japan on March 11, 2011, and the subsequent nuclear accident at Tokyo Electric Power Company's Fukushima Dai-ichi nuclear power plant have resulted in worldwide attention toward improving nuclear safety.

EPRI, INPO, and the Nuclear Energy Institute (NEI), in conjunction with senior utility executives, created a joint leadership model to integrate and coordinate the U.S.-nuclear industry's response to events at the Fukushima Dai-ichi nuclear energy facility. This model will ensure that lessons learned are identified and well understood, and that response actions are effectively coordinated and implemented throughout the industry.

America's nuclear energy industry is taking action based on a preliminary understanding of the events. The industry's response is structured to ensure that emergency response strategies are updated based on new information and insights learned during subsequent event reviews.

Separately, the NRC is conducting an independent assessment and will consider actions to ensure that its regulations reflect lessons learned from the Fukushima events.

The primary objective of the response is to improve nuclear safety by learning and applying the lessons from the Fukushima Dai-ichi nuclear accident. In response, the U.S. nuclear industry has established the following strategic goals to maintain and provide, where necessary, added defense in depth for critical safety functions, such as reactor core cooling, spent fuel storage pool cooling, and containment integrity:

- The nuclear workforce remains focused on safety and operational excellence at all plants, particularly because of the increased work that the response to the Fukushima event will represent.

- Timelines for emergency response capability to ensure continued core cooling, containment integrity and spent fuel storage pool cooling are synchronized to preclude fuel damage following station blackout (SBO) or challenges to the ultimate heat sink.

- The U.S. nuclear industry is capable of responding effectively to any significant event in the United States with a scalable response to support an international event, as appropriate.

- Severe accident management guidelines, security response strategies, and external event response plans are effectively integrated to ensure that nuclear energy facilities can provide a symptom-based response to events that could affect multiple reactors at a single site.

- Margins for protection from external events are sufficient based on the latest hazards analyses and historical data.

- Spent fuel pool (SFP) cooling and makeup functions are fully protective during periods of high heat load in the SFP and during extended SBO conditions.

- Primary containment protective strategies can effectively manage and mitigate postaccident conditions, including elevated pressure and hydrogen concentrations.

- Accident response procedures provide steps for controlling, monitoring and assessing potential radiation and ingestion pathways during and following an accident, including timely communication of accurate information.

The industry has established principles to guide the development of its response actions. These principles will be used to guide the resolution of issues and plant improvements and will ensure that a consistent expectation is established for incorporating lessons into the operations at each site. The strategic response actions will be designed to do the following:

- Ensure that equipment and guidance, enhanced as appropriate, result in improvements in response effectiveness, using a diverse, flexible and performance-based approach for beyond-design-bases activities.

- Address guidance, equipment and training to ensure long-term viability of safety improvements.

- Develop performance-based, risk-informed response strategies that account for unique site characteristics.

- Maintain a strong interface with Federal regulators to ensure that regulatory actions are consistent with safety significance and that compliance can be achieved in an efficient manner.

- Coordinate with Federal, State and local Government and their emergency response organizations on industry actions to improve overall emergency response effectiveness.

- Aggressively communicate the forthright approach that the U.S.-industry is taking to implement the lessons learned from the Fukushima Dai-ichi accident.

The following seven building blocks, shown with the lead organization(s), form the core strategy for the American response to the accident at Fukushima:

(1) Maintain Focus on Excellence in Existing Plant Performance (INPO). Focus on continued performance improvement of U.S. reactors.

(2) Develop and Issue Lessons Learned from the Fukushima Events (INPO). Focus on a comprehensive analysis of the Fukushima event and show that lessons learned are applied to the U.S. nuclear industry and shared with WANO.

(3) Improve the Effectiveness of U.S. Industry Response Capability to Global Nuclear Events (INPO/NEI). Focus on identified lessons learned from the U.S. industry response to the Fukushima event and allow for a more effective integrated response to future events.

(4) Develop and Implement a Strategic Communications Plan (NEI). Focus on managing the industry's strategic communications and outreach campaigns to recover policymaker and public support for nuclear energy.

(5) Develop and Implement the Industry's Regulatory Response (NEI). Focus on managing the industry's regulatory interactions and resolution of applicable industry regulatory issues from the incident.

(6) Participate and Coordinate with International Organizations (INPO/EPRI). Focus on ensuring that the results from international investigations are captured and effectively used to inform actions with the other building blocks.

(7) Provide Technical Support and Research and Development Coordination (EPRI/NSSS Owners' Groups). Focus on existing technical solutions and research and development activities and deliverables necessary to address recommended actions of this plan.

Each building block is supported by nuclear and, in specific instances, nonnuclear industry organizations and companies that require specific technical, operational or other expertise

In addition to directly supporting "the way forward" response to the Fukushima accident, INPO has issued the following three INPO event reports (IERs) on Fukushima, as summarized:

(1) IER 11-1, "Fukushima Dai-ichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami" dated March 15, 2011, and its supplement dated October 3, 2011

The events at the Fukushima Dai-ichi plant appear to be caused by factors directly affecting nuclear safety that are outside the design-basis for the facility. Although details on the full extent of damage to these units remain unknown, these factors represent a significant challenge to the nuclear safety of these units. Immediate actions by the U.S. industry are appropriate to assess and take corrective actions to address potential vulnerabilities that could challenge response to events that are beyond site design bases.

The following four recommendations were provided to the U.S. nuclear industry to provide near-term assurance that each station is in a high state of readiness to respond to both design-basis and beyond-design-basis events:

1. Verify the capability to mitigate conditions that result from beyond design basis events, typically bounded by security threats, committed to as part of Section B.5.b of the NRC Security Order, dated February 25, 2002, and

severe accident management guidelines. Include, but do not limit, the verification to the following:

- Verify through test or inspection that equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected. (The intent is not to retest permanently installed equipment that is tested under a regulatory testing regime.)
- Verify through walkdowns or demonstration that procedures to implement the above strategies are in place and are executable. (The intent is not to connect to, or operate, permanently installed equipment.)
- Verify that the qualifications of operators and the support staff needed to implement the procedures and work instructions are current.
- Verify that any applicable agreements and contracts are in place and can meet the conditions needed to mitigate the consequences of these events.

2. Verify that the capability to mitigate SBO conditions required by station design is functional and valid, as fallows:

    - Verify through walkdowns and inspection that all required materials are adequate and properly staged.
    - Demonstrate through walkdowns that procedures for response to an SBO are executable.

3. Verify the capability to mitigate internal and external flooding events required by station design, as follows:

    - Verify through walkdowns and inspections that all required materials and equipment are adequate and properly staged. These walkdowns and inspections shall include verification that accessible doors, barriers, and penetration seals are functional.

4. Perform walkdowns and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment's function could be lost during seismic events appropriate for the site. Develop mitigating strategies for identified vulnerabilities. As a minimum, perform walkdowns and inspection of important equipment (permanent and temporary), such as storage tanks, plant water intake structures, and fire and flood response equipment, and develop mitigating strategies to cope with the loss of that important function.

(2) IER 11-2, "Fukushima Dai-ichi Nuclear Station Spent Fuel Pool Loss of Cooling and Makeup," dated April 25, 2011

The earthquake and tsunami caused the loss of all station power supplies and are believed to have damaged some of the SFP cooling water systems. As a result, all SFP cooling and makeup was lost to each of the Fukushima Dai-ichi SFPs.

The loss of cooling to the fuel pools for Units 1, 2, 3, and 4 resulted in the pools heating up and ultimately reaching saturation or near saturation temperatures. The resultant evaporation reduced the SFP inventories.

The inability to maintain SFP water inventory in multiple units resulted in extraordinary recovery efforts. These actions included helicopter seawater drops, fire truck seawater sprays, water cannons, and fire pump seawater injection into the fuel pool cooling systems. Recovery efforts and operator access to the SFPs were limited by adverse conditions, including high dose rates, radiological contamination, and reactor building and plant systems damage. A lack of recovery plans and suitable makeup equipment is believed to have further hindered fuel pool cooling and water inventory recovery.

The loss of SFP cooling, coolant inventory, and makeup capability at the Fukushima Dai-ichi plant may have resulted in damage to stored spent nuclear fuel and significant radiological consequences to station personnel, the site, and the surrounding region. The event was caused by factors that were outside the design basis for the facility.

The following five recommendations were provided to the U.S. nuclear industry to ensure that each station will increase its sensitivity to spent fuel storage event response and that each station will maintain a high state of readiness to respond to events that challenge spent fuel storage integrity:

1. For outage periods, verify the implementation of actions to address Recommendations 1 - 4 and Recommendations 6 - 12 in SOER 09-1, "Shutdown Safety," as they relate to the safety functions associated with SFP cooling and inventory makeup. Implement this recommended action within 60 days.

2. For online periods when the time for the SFP to reach 200 degrees Fahrenheit upon loss of normal cooling is less than 72 hours, establish controls to identify and protect systems and equipment required to maintain the functions of SFP decay heat removal and inventory control. The controls should include the following:

    - Protected systems and equipment are clearly identified in the field to prevent inadvertent work on or near protected equipment. Physical barriers are used whenever possible, particularly in areas in which personnel could bump into a component, thereby causing an inadvertent trip or system transient. Protected spaces are monitored to ensure that barriers are in place and that unauthorized work is not occurring. Nonintrusive work is controlled and is limited to activities, such as visual inspections and operator rounds.
    - For work required on protected SFP equipment, support systems, or backup equipment, establish specific management controls for the conduct of work. These controls will include additional barriers, such as walkthroughs, contingencies, and direct management oversight. Establish compensatory actions for the SFP decay heat removal and inventory control functions commensurate with the risk of the associated SFP configuration. The establishment of compensatory

actions will prevent the SFP from reaching saturation conditions on loss of cooling.

3. For all plant conditions, establish the time for the SFP to reach 200 degrees Fahrenheit (bulk temperature) in the event that normal cooling is lost. Maintain this information in a format that is readily available in the control room and emergency response facilities. This time is intended for information purposes only in case a sustained loss of SFP cooling or inventory occurs. Implement this recommended action within 90 days.

4. Verify the adequacy of station abnormal operating procedures for responding to the loss of SFP cooling or inventory, or both. Ensure that these procedures include actions and contingencies to monitor SFP level and temperature, and include the capability to make up inventory to the SFPs during a loss of all alternating current (ac) power. Verify that the guidance in the abnormal operating procedure can be implemented during severe weather, seismic events, loss of control room, and flood conditions.

5. Revise station emergency operating procedures to include a precautionary statement that SFP level and temperature should be monitored.

(3) IER 11-4, "Near-Term Actions to Address the Effects of an Extended Loss of All ac Power in Response to the Fukushima Dai-ichi Event," dated August 1, 2011

Some earthquake accelerations and the large tsunami that struck the Fukushima Dai-ichi site exceeded the design basis for the site. The tsunami inundated the area around Units 1 - 6, causing extensive damage to site buildings and switchyards and flooding in the turbine and reactor buildings. Intake structures were damaged, and a loss of all seawater cooling pumps that provide equipment cooling occurred. The diesel generators started and ran; however, the combination of a loss of cooling water, flooding of electrical switchgear, and flooding of some of the diesel generator rooms installed in the basement of the turbine buildings caused a loss of all ac power on site for Units 1 - 4.

Flooding from the tsunami caused a complete loss of battery power (direct current (dc) power) at Unit 1 and Unit 2 and partial losses of dc power at Unit 3, which further complicated the event response. The extended loss of all ac power and dc power as the batteries were depleted exceeded the ability of the units to maintain essential equipment functions, thus resulting in eventual fuel damage in Units 1, 2, and 3.

In the days that immediately followed the events, equipment, operational, and organization challenges further complicated the event response. Operators and other emergency response personnel were faced with mitigating the event that affected multiple units with extensive damage to plant systems and infrastructure and that therefore went beyond predetermined procedures and the capability of available portable equipment. The station was equipped with several sources of emergency reserve power. Each unit was equipped with offsite emergency power, EDGs, and two emergency dc power trains for critical equipment. Within about 12 hours following the tsunami, portable emergency generators were brought to the site; however, operators were unable to power key equipment because the cable connections were not compatible with plant equipment and because the emergency power and the electrical distribution equipment were unavailable because of flooding. Emergency response personnel identified alternate means for powering key equipment, including retrieving car batteries when all emergency response batteries were depleted to power key instruments for operator monitoring and control of critical safety functions. Later in event response, emergency diesel-driven pumping and ac power systems were used to provide cooling and makeup to the reactors and for fuel stored in SFPs.

Restoring power to installed electrical equipment was delayed because support equipment and large portions of the electrical distribution system, roadways, and structures were damaged. Power was restored to many components and was ultimately successful when personnel bypassed large portions of the electrical distribution system. The inability to provide necessary power to permanent equipment and the delays in powering temporary equipment needed to establish core cooling directly caused fuel damage during the event.

Most U.S. plants have 4-hour coping durations for mitigating SBO conditions. U.S.-plants also developed emergency response strategies to mitigate the effects of fires postulated to adversely affect safety system functions. In many cases, stations rely on SBO diesel generators, gas turbines, or ac power from other onsite sources to mitigate the blackout condition. Although existing capabilities for coping with loss of ac power conditions are robust, postulating low-probability events and scenarios that are beyond SBO design basis and that challenge those capabilities is possible. IER 11-4 details processes by which station personnel can identify reasonable strategies and actions to extend the time in which existing equipment can be used to maintain critical safety functions for extended loss of ac power until additional equipment can be supplied to support long-term safe shutdown conditions. The scope of this effort should include operating conditions to determine the most limiting conditions.

In addition to increasing operating margin for loss-of-power events, the increased coping times resulting from this effort will be used by industry groups that are developing strategies for emergency response. The implementation of actions that extend this time and that are permitted by current license conditions should be done on a priority basis.

The recommendations of IER 11-4 are twofold. First, the report calls for the development of preplanned contingencies for protection from the extended loss of ac power and beyond-SBO events, similar to those experienced at Fukushima Dai-ichi pending longer-term industry response. Second, the report requires stations to provide unit-specific information on coping time and design limitations for extended loss of power events to support U.S. industry awareness and response to the Fukushima Dai-ichi event.

Actions that improve operating margin for a beyond SBO event that can be accomplished within existing design and licensed conditions should be implemented. Actions, including those that would require a license amendment, should be identified and included with the station response to IER 11-4. If modifications are considered, they should reduce installation time for temporary hookups. (For example, installed/flanged and electrical connections are preferred over the need for the disassembly of a valve, pipe, or breaker panel). A supplement to IER 11-4 will be issued at a later date with additional recommended actions based on industry input. These additional actions should provide guidance for specific mitigating strategies to further increase margins of safety for the extended loss of ac power events. Industry working group meetings are planned to support the development of common industry approaches for addressing these recommendations.

The following four specific recommendations were provided to the U.S. industry:

1. For all units, develop methods to maintain (or restore) core cooling, containment integrity, and SFP inventory using existing installed and portable equipment during an extended loss of electrical ac power event that lasts at least 24 hours. This recommendation includes implementing actions to address the loss of ac power events (beyond SBO) simultaneously at each unit of multi-unit sites using the conditions described above. Implement actions to improve operating margin that can be accomplished within the existing license, as follows:

    - Report the length of time the station can maintain the critical safety functions listed in Recommendation 1 using existing installed equipment, even if the duration of time is less than 24 hours. Identify and report conditions that limit the achievement of the 24-hour duration.
    - Report the length of time the station can maintain the critical safety functions listed in Recommendation 1 using existing installed and portable equipment, even if the duration of time less than 24 hours. Identify and report conditions that limit the achievement of the 24-hour duration or longer durations. Describe and report the protective measures or the measures that differentiate portable equipment from installed electrical ac power sources.
    - If enhancements or station upgrades are proposed for extending the station's ability to increase operating margin for extended the loss of ac power events, include the proposed upgrades and the expected margin improvement in the response. The proposed upgrades will inform the industry working group process on methods and strategies recommended for broad industry consideration.

2. Identify essential instrumentation needed for monitoring core, containment, and spent fuel safety. Develop methods to ensure these functions are maintained throughout an extended loss of ac power event. This recommendation includes the conduct of a plant-specific analysis of methods that would be used. Specifically, methods and instructions should include the identification of needed equipment and materials to power the minimum essential components in the event that installed dc batteries are depleted.

3. Develop methods for providing fuel to power emergency response equipment. Develop strategies for obtaining fuel oil sufficient to operate temporary power equipment in the event of a loss of all site ac power sources that lasts at least 24 hours. Onsite fuel oil reserves that are protected from flood and seismic events appropriate for the site may be credited for this recommendation.

4. Provide communications equipment suitable for onsite and offsite communication needs during an extended loss of ac power event. Develop a means for communicating with emergency response personnel for an extended loss of ac power event. To assess a needed communications strategy, expect that ac power is not available to cell phones or other communication infrastructures within 25 miles of the plant site.

INPO will followup on the status of the IER recommendations during its review process in 2013 and 2014.

Diverse and Flexible Coping Strategies (FLEX)

Working with the United States nuclear industry, NEI and INPO developed a "Diverse and Flexible Coping Strategy" that was endorsed by the NRC in August 2012. It provides a diverse and flexible means to prevent fuel damage while maintaining the containment function in beyond design basis external event conditions resulting in an:

- Extended loss of ac power, and
- Loss of normal access to the ultimate heat sink

The objective is to establish an essentially indefinite coping capability by relying upon installed equipment, onsite portable equipment, and prestaged offsite resources. FLEX employs a three-phase approach:

- Phase 1: following the event and prior to the time when portable equipment can be deployed, the plant must be able to maintain the key safety functions using installed equipment.
- Phase 2: with adequate time and staffing, onsite portable equipment is deployed.
- Phase 3: after 24 hours, offsite equipment can be deployed to sustain key safety functions indefinitely.

In summary, the concept is diverse and flexible to enable deployment of the strategies for a range of initiating events and plant conditions.

The offsite staged equipment strategy consists of the following elements:

- list off-site equipment needs
- base on realistic natural disasters, such as floods and tornados
- leverage current inventories
- standardize interconnections
- alignment with onsite coping strategies
- determine required deployment times
- determine offsite locations
- determine logistics, transport, and shipping requirements
- establish sharing agreements
- plan for self-sufficiency, but include government

The concept for offsite support is based on the assumption that onsite resources must be sufficient to cope for the first 24 hours. FLEX analyses will determine what coping equipment can be credited as coming from offsite sources. Procedures used to respond to FLEX will address contacting the offsite sources. A standardized list of equipment connectors is being developed to address interchangeability of the equipment. Each site is required to have N+1 sets of FLEX equipment onsite to respond to the event. Therefore, these sites become a source of FLEX equipment for a site in such an event. The United States nuclear industry has been sharing parts information since 1990 via a system that is routinely used by all of the plants. Over 153,000 searches for parts have been conducted this year. During an emergency event, a call to INPO or directly to the other site will activate mobilization of FLEX equipment from other sites.

In addition to support from other sites, there will be two regional response centers, in Memphis and Phoenix, capable of delivering equipment to any site. The regional response centers will be managed by a vendor, Strategic Alliance for FLEX Emergency Response. The Pooled Equipment Inventory Company has joined forces with AREVA to create the Strategic Alliance for FLEX Emergency Response team to develop and manage the regional response center program as part of the Pooled Equipment Inventory Company's existing Pooled Inventory Management Program for the United States nuclear industry. Each regional response center will have five sets of FLEX equipment: 4 sets to support sites and 1 set out of service for maintenance. In addition, each regional response center will have any additional equipment specified by a site in their site-specific regional response center mobilization manual.

Each site will identify a staging area for delivery of the equipment. The regional response center will deliver the specified equipment to the staging area within 24 hours of being notified. Delivery will be by air or ground depending on the distance of the site from the center. Support will include equipment and personnel. Equipment technicians will accompany the equipment to the site to assist with set up and deployment. Qualified technicians from the 63 other facilities can be dispatched.

The timeline for emergency response is as follows:

<u>24 Hour FLEX Equipment</u>

- T-0: notification
- T-2: mobilization
- T-4: transportation to staging area
- T-20: equipment preparation
- T-22: transportation to site
- T-24: first equipment on-site

<u>72 Hour FLEX Equipment</u>

- All plants: FLEX equipment to be delivered by ground transportation

11. Conclusion

The U.S. commercial nuclear industry has made substantial, sustained and quantifiable improvement in plant safety and performance during the 3 decades since the Three Mile Island event. The leaders who guided this industry over decades of challenge and change showed great insight when they recognized the need for an unprecedented form of industry self-regulation through peer review. The industry members acknowledged that nuclear energy would remain a viable form of electric power generation only if utilities could ensure the highest levels of nuclear safety and reliability (i.e., the achievement of excellence) in nuclear power plants. The industry responded to this challenge by creating an independent oversight process of the highest integrity and by requiring of itself an uncompromising commitment to the standards and ethical principles that are essential to success.

This insight and commitment to integrity has provided the foundation for a unique, sustained partnership between INPO and its members. INPO is pleased to serve as an essential element of an industry that has raised its standards and improved its performance in nearly every aspect of plant operation. INPO does not take credit for this success; however, it does take pride in its contribution to that success.

INPO also recognizes that the pursuit of excellence is a continuing journey, not a destination. As the U.S.-nuclear industry evolves and advances, it will continue to encounter situations that challenge both people and equipment in a business environment that is competitive, complex, and increasingly global in character.

These challenges, although demanding, are not insurmountable. The U.S. commercial nuclear industry, in partnership with INPO, will continue the tradition of both sharing insight and acting with integrity and, in doing so, will continue on the shared journey to ever higher levels of excellence.

APPENDIX A
NRC STRATEGIC PLAN 2008 - 2013

The U.S. Nuclear Regulatory Commission (NRC) published NUREG-1614, Volume 5, "Strategic Plan: Fiscal Years 2008–2013 (Updated)" in February 2012. Appendix A to this report summarizes the key points of this plan.

Future Challenges

Reviewing applications to construct and operate new nuclear power plants (including small modular reactors) while continuing to ensure the safe and secure operation of the existing licensed facilities, and addressing any national policy decisions related to the management of radioactive waste are major evolving challenges facing the NRC over the next several years.

To meet these challenges, the NRC must use its resources efficiently, revise the regulatory framework as appropriate to disposition existing or emerging issues, and provide an adequate infrastructure to maintain staff competence and readiness. Even as the NRC works to address these challenges, the agency's mission and values remain unchanged. The NRC's priority continues to be ensuring the adequate protection of public health and safety, and promoting the common defense and security. Safety and security remain the agency's core functions upon which the goals and strategic outcomes of this strategic plan are based. This focus on safety and security ensures that the NRC remains a strong, independent, stable, and effective regulator.

In the next 2 years, the NRC expects to do the following:

- Receive additional applications for new uses of radioactive material and applications from entities that want to build and operate both small and large new nuclear power plants. In addition, the agency will develop the regulatory infrastructure to support the review of anticipated applications for small modular reactors.

- Anticipate an increase in the quantities of spent nuclear fuel that will be held in interim storage at reactor sites or possibly transported to centralized interim storage sites to await permanent disposal.

- Coordinate with a wide array of Federal, State, local, and Tribal Governments on matters related to nuclear material regulation, license renewal, new reactor licensing, homeland security, emergency planning, management of radioactive waste, decommissioning, and environmental protection.

- Make additional improvements in the agency's regulatory system based on lessons learned from the nuclear accident that began with the events on March 11, 2011, at the Fukushima Dai-ichi nuclear facility in Japan.

- Increase international engagement on safety and security of the use of nuclear material.

The NRC recognizes that the developments noted above will create an even greater need for effective and open communication with public stakeholders on a variety of issues. These issues include emergency preparedness and the safety and security of existing and proposed nuclear power plants; fuel cycle facilities; and medical, academic, and industrial uses of licensed materials.

The resolution of these complex regulatory issues also requires effective knowledge management to capture, retain, and leverage institutional knowledge. The agency will continue to attract staff with the skills to conduct complex safety reviews. To retain these highly skilled and educated professionals, the agency realizes that they will need effective leadership, access to the tools to perform their jobs, and a workplace that promotes strong employee engagement. The NRC's approach focuses on ensuring that each staff member is highly trained in the skills related to his or her duties; the regulatory processes that govern agency actions; and the regulatory principles inherent in making the agency a strong, independent, stable, and predictable regulator.

Being a stable and predictable regulator means having effective and structured regulatory processes in place and ensuring that these processes are followed. The NRC will develop regulatory initiatives in a manner that is open to public review and involvement in accordance with these processes. The NRC is committed to considering, and being responsive to, stakeholder input before implementing new regulatory initiatives.

Key External Factors

The NRC's ability to achieve its goals depends on a changing mix of industry operating experience, national priorities, legislation, market forces, and resource availability. The NRC will continue to refine and implement processes for managing change to ensure that the agency is ready to address shifting priorities in a timely manner. The following information discusses significant external factors that, although they are beyond the control of the NRC, could still affect the agency's ability to achieve its strategic goals:

- Significant Operating Incident (Domestic or International). A significant incident at a nuclear facility could cause the agency to reassess its safety and security requirements, which could change the agency's focus on some initiatives related to its goals until the situation stabilizes. Because NRC stakeholders (including the public) are highly sensitive to many issues concerning the use of radioactive materials, even events of relatively minor safety or security significance could potentially require a response that consumes considerable agency resources.

- Significant Terrorist Incident. A significant terrorist incident anywhere in the United States would heighten the NRC's oversight and response stance. Subsequent new or changed security requirements or other policy decisions might affect the NRC, its partners, and the industry it regulates. A significant terrorist incident at a nuclear facility or activity anywhere in the world that departs from the agency's evaluation of threat parameters could affect the NRC's priorities and could potentially affect U.S. policy for export activities, the NRC's role in international security, and requirements for security at U.S. nuclear power plants and other licensee facilities.

- Emergency Preparedness and Incident Response. Emergency preparedness and incident response activities with Federal, State, and local agencies and with Tribal

Governments continue to increase in scope and number. These activities affect the agency's priorities and workloads.

- Legislative Initiatives. Legislative initiatives under consideration by Congress can have a major impact on the NRC. For example, initiatives concerning cyber security may potentially affect the NRC's regulatory framework.

- International Nuclear Safety Developments. The international community, often through the International Atomic Energy Agency (IAEA), changes and updates international standards that could affect the NRC. For instance, the IAEA could develop new standards or change current standards, and the NRC would then need to address stakeholder calls (both domestically and internationally) to implement the new standards for U.S. licensees. The NRC will need to actively engage with the international community, including the IAEA, multilaterally, and bilaterally.

- International Treaties and Conventions. The ratification of international instruments on safety and security in the sector of nuclear materials and facilities imposes binding provisions on the Nation and its corresponding Governmental agencies, such as the NRC and the U.S. Department of Energy. These obligations and the resources necessary to ensure compliance with them compete with all the other programs and activities that are within the role and responsibility of the NRC as the national regulatory body.

- National Strategy on Nuclear Waste. The development of a potential national strategy for the management of high-level waste and spent nuclear fuel will continue to present challenges in setting the strategic direction of the agency's spent fuel and high-level waste management programs. The Nation will need to consider the recommendations of the President's Blue Ribbon Commission on America's Nuclear Future, which was released in January 2012, in setting the direction of associated agency programs related to nuclear waste storage, transport, disposal, and reprocessing.

APPENDIX B
NRC MAJOR MANAGEMENT CHALLENGES FOR THE FUTURE

By law, the Inspector General of each Federal agency (discussed in Article 8 of Part 2 to this report) must describe what he or she considers to be the most serious management and performance challenges facing the agency and must assess the agency's progress in addressing those challenges. Accordingly, the Inspector General of the U.S. Nuclear Regulatory Commission (NRC) prepared his annual assessment of the major management challenges confronting the agency. The NRC published the latest report in October 2012; this report can be found on the agency's public Web site.

In his assessment, the Inspector General defined serious management challenges as "mission-critical areas or programs that have the potential for a perennial weakness or vulnerability that, without substantial management attention, would seriously impact agency operations or strategic goals." The challenges identified represent critical areas or difficult tasks that warrant high-level management attention. In the 2012 report, the Inspector General identified the seven management challenges describe below to be the most serious as of October 1, 2012.

Challenge 1: Management of Regulatory Processes To Meet a Changing Environment in the Oversight of Nuclear Materials

The NRC is responsible for maintaining an established regulatory framework for the safe and secure use of nuclear materials; medical, industrial, and academic applications; uranium recovery activities; and low-level radioactive waste sites. The NRC is authorized to grant licenses for the possession and use of radioactive materials and establish regulations to govern the possession and use of those materials. Agency regulations require that certain material licensees to have extensive material control and accounting programs as a condition of their licenses. Other license applicants (including those requesting authorization to possess small quantities of special nuclear materials) must develop and implement plans that demonstrate a commitment to accurately control and account for radioactive materials. Upon a State's request, the NRC may enter into an agreement to relinquish its authority to the State to regulate certain radioactive materials and limited quantities of special nuclear material. The State must demonstrate that its regulatory program is adequate to protect public health and safety and that is compatible with the NRC's program. The States that enter into an agreement assuming this regulatory authority from NRC are called Agreement States. Currently, there are 37 Agreement States.

Challenge 2: Management of Internal NRC Security and Oversight of Licensee Security Programs

The NRC must remain vigilant of the security of its own infrastructure and that of nuclear facilities and nuclear material. Ensuring predictability in the security environment is an ongoing challenge for the NRC. The agency must continue to use robust, proactive measures to protect its infrastructure – the buildings, personnel, and information – from both internal and external threats. Moreover, as the nature of the threat continues to evolve, the NRC faces challenges with the protection of nuclear facilities and materials, the constant sharing of sensitive information, and emergency preparedness and incident response.

Challenge 3: Management of Regulatory Processes to Meet a Changing Environment in the Oversight of Nuclear Facilities

The NRC faces the challenge of maintaining its core regulatory programs while adapting to changes in its regulatory environment. The NRC must address a highly variable interest in the licensing and construction of new nuclear power plants to meet the Nation's increasing demands for energy production. As of July 2012, the NRC had received 18 combined license applications, 10 of which the NRC was actively reviewing. Moreover, the agency is reviewing two standard design certifications and, for advanced reactors, expects to receive two design certification applications and one construction permit application through 2014.

While responding to the emerging demands associated with licensing and regulating new reactors, the NRC must maintain focus and must effectively carry out its current regulatory responsibilities, such as inspections of the current fleet of operating nuclear reactors and fuel cycle facilities. The NRC intends to increase its safety focus on licensing and oversight activities through risk-informed and performance-based regulation.

Furthermore, in June 2012, the U.S. Court of Appeals for the District of Columbia Circuit ruled that the NRC's waste-confidence decision had not adequately addressed all environmental effects and that it had therefore violated the National Environmental Policy Act. In September 2012, the Commission directed the NRC staff to develop an environmental impact statement, a revised waste confidence decision, and a rule on the temporary storage of spent nuclear fuel -- all within the next 24 months. Although the NRC staff believes that the court ruling will not delay the license application review activities and ongoing plant construction, its ultimate impact on final licensing decisions and new construction approvals has not yet been determined.

Challenge 4: Management of Issues Associated with the Safe Storage of High-Level Radioactive Waste When There Is No Long-Term Disposal Solution

The NRC regulates high-level radioactive waste generated from commercial nuclear power reactors. High-level radioactive waste is either spent (used) reactor fuel when it is accepted for disposal or the waste materials that remain after the spent fuel is reprocessed. Because of its highly radioactive fission products, high-level radioactive waste must be handled and stored with care. Because the only way radioactive waste finally becomes harmless is through decay, which, for high-level waste can take hundreds of thousands of years, the waste must be stored and finally disposed of in a way that adequately protects the public for a very long time.

The United States has entered a period in which the national policy for the storage, reprocessing, and disposal of spent nuclear fuel is being reexamined. Spent nuclear fuel may be stored at reactor sites for the foreseeable future because of the uncertainty surrounding a permanent repository for high-level radioactive waste. As such, the NRC has been reviewing the issues associated with long-term storage. An independent spent fuel storage installation is an NRC–licensed facility designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with the spent fuel. An independent spent fuel storage installation typically consists of a concrete storage pad, storage containers (casks), and any support facilities. As of August 2012, independent spent fuel storage installations at 62 different locations across the United States were storing spent nuclear fuel or were preparing to store spent nuclear fuel in the near term. Of these 62 independent spent fuel storage installation sites, 52 were located at operating reactors, and the remaining 10 were located away from an operating reactor.

In 2010, the NRC updated its waste confidence decision to affirm that spent nuclear fuel could be safely stored onsite at nuclear power plants until a permanent waste repository is built. However, in June 2012, the U.S. Court of Appeals for the District of Columbia Circuit ruled that the NRC's waste-confidence decision had not adequately addressed all environmental effects and that it therefore violated the National Environmental Policy Act. The Commission affirmed that the agency will not issue licenses that depend on the waste confidence decision until it has appropriately addressed the ruling by the U.S. Court of Appeals for the District of Columbia Circuit.

Challenge 5: Management of Information Technology

The NRC needs to continue upgrading and modernizing its information technology capabilities to meet its information technology/information management strategic goals. These goals include (1) ensuring that the NRC staff has quick and easy access to information, (2) providing information technology solutions that are easy to use and that increase agency program performance, and (3) delivering excellent service.

Challenge 6: Administration of All Aspects of Financial Management and Procurement

NRC management is responsible for meeting the objectives of several statutes, including the Federal Managers Financial Integrity Act of 1982. This Act mandates the NRC to establish controls that reasonably ensure that (1) obligations and costs comply with applicable law, (2) assets are safeguarded against waste, loss, unauthorized use, or misappropriation, and (3) revenues and expenditures are properly recorded and accounted for accordingly. This Act encompasses program, operational, and administrative areas and accounting and financial management.

The NRC's procurement of goods and services must be made with an aim to achieve the best value for the agency's dollars in a timely manner. Agency policy states that the NRC's procurement of goods and services should support the agency's mission; be planned, awarded, and administered efficiently and effectively; and should be consistent with sound business practices and contracting principles. The agency is currently focusing its efforts on achieving (1) a 21st century acquisition program that uses state-of-the-art acquisition methodologies for acquisition planning, execution, management, and closeout and (2) an acquisition program that fully integrates with agencywide program and financial planning and execution.

Challenge 7: Management of Human Capital

For several years, the NRC experienced significant growth because of the increased interest in nuclear power. During fiscal year 2012, the NRC's workforce was approximately 4,000 staff positions; the agency will probably not see any growth over the next several years. Going forward, the NRC will need to support increasing mandates within a zero-growth or declining budget environment. The NRC must institutionalize an approach that focuses on its mission of protecting the public health and safety while remaining mindful of staff needs. To manage human capital effectively while accomplishing the agency's mission, the NRC must continue to: (1) reduce inefficiencies and overhead by centralizing and streamlining processes while maintaining or improving the level of customer service and (2) implement space planning.

APPENDIX C
U.S. SUPPORT OF THE INTERNATIONAL ATOMIC ENERGY AGENCY ACTION PLAN ON NUCLEAR SAFETY

The United States has strongly supported the International Atomic Energy Agency (IAEA) as it has identified specific initiatives to address lessons learned from the Fukushima Dai-ichi nuclear accident and to enhance multilateral communication, including identifying new resources or redistributing existing ones. In each of these areas, the United States has emphasized the importance of close cooperation among parties to maximize the effectiveness of the initiatives and to avoid duplication of efforts.

The U.S. Government participated in the development of IAEA's report entitled, "Action Plan on Nuclear Safety," which was approved by the Board of Governors and adopted by the General Conference in September 2011. Before the plan's approval, the U.S. Government participated in major international meetings on the Fukushima Dai-ichi accident in 2011, including meetings hosted by the French Government as 2011 President of the Group of Eight Industrialized Nations (G8) and parallel meetings of the Group of Twenty (G20) in Paris, the Nuclear Energy Agency (NEA) forum in Paris, the IAEA ministerial-level conference in Vienna, and the European Nuclear Safety Regulators Group meeting in Brussels. U.S. Nuclear Regulatory Commission (NRC) experts also participated in several IAEA international peer review missions to Japan, including the initial fact-finding mission that took place in May 2011.

The examples described below are some actions taken by the United States in support of the IAEA's Action Plan.

Safety Assessments in Light of the Accident at Tokyo Electric Power Company's Fukushima Dai-ichi Nuclear Power Station

The United States immediately undertook a comprehensive assessment of its 104 operating nuclear power plants (the outcomes of which are explained in detail in Section 19 of this report).

The NRC played a leadership role in the NEA Senior Task Group on the Impacts of the Fukushima Accident and in ongoing work within the NEA committee structure. The NRC informed the task group of its efforts in response to the Fukushima accident; in turn, the NRC was informed of the activities of the other countries. The NRC encouraged the task group to work with the IAEA and to coordinate with other international activities. In addition, the NRC is currently playing a leadership role on the newly formed Task Group on Accident Management. The Fukushima task group recommended creation of the Task Group on Accident Management. The group is focusing on the regulatory aspects of accident management by the licensee on site, and it expects to complete its work in December 2013.

IAEA Peer Reviews

The United States has strongly supported the IAEA's suite of peer review services since their inception. The NRC regularly provides technical experts to participate in Integrated Regulatory Review Service (IRRS) and Operational Safety Review Team (OSART) missions around the world, often at a senior leadership level. The United States regularly hosts OSART missions at U.S. nuclear power plants and hosted an IRRS mission focused on the U.S. operating reactor program in 2010. A followup IRRS mission is scheduled for early 2014 for which the team will review the NRC's implementation of recommendations and best practices. In October 2011, the NRC hosted an international workshop on lessons learned from IRRS missions, which discussed the potential impacts of the Fukushima accident on nuclear power programs in accordance with the direction of the IAEA's Action Plan on Nuclear Safety to enhance the international peer review process. In addition, the NRC is participating in the IAEA's ongoing effort to strengthen the IRRS program through modification of guidance documents to enable the IRRS teams to be more focused on priority safety challenges. For the past 4 years, the United States has funded a cost-free expert in the IAEA's Division of Nuclear Installation Safety, Regulatory Activities Section, in support of the IRRS program and other peer review activities, such as the International Nuclear Infrastructure Review. This cost free expert has helped IAEA to make its programs more effective and efficient.

Emergency Preparedness and Response

As discussed in Section 16 of this report, the United States has undertaken significant activities to assess and strengthen, where appropriate, its emergency preparedness and response programs. The United States has also worked closely with Canada and Mexico to enhance North American cooperation in this area. The NRC, through its member on the International Nuclear and Radiological Event Scale Advisory Committee, is developing additional guidance on the application of the international event scale in severe accidents. These activities, both separately and cumulatively, streamline and prioritize response protocols that enhance public confidence.

National Regulatory Bodies

As discussed in the section entitled, "IAEA Peer Reviews," above, the NRC has devoted significant resources to address the findings and recommendations from the 2010 U.S. IRRS mission report and has scheduled a followup mission for early 2014. In addition, as discussed in detail in Section 8.1.5.3 of this report, a thorough assessment of the U.S. nuclear safety regulatory infrastructure and current regulations was a key component in the NRC's assessment methodology. The results of the NRC's assessment indicated that the agency's independence as a regulatory body and its regulations for nuclear power plants were sufficient. The assessment identified numerous areas in which improvements were necessary to ensure a continued high level of safety in the United States.

Operating Organizations

As discussed in Section 9 of this report, the licensee has primary responsibility for safety in the United States. Since the Fukushima Dai-ichi accident, the NRC has worked to ensure clear communication with each of its licensees and has directed each licensee to implement recommendations from nuclear safety assessments. The U.S. Government continues to work closely with the Institute for Nuclear Power Operations (INPO) to ensure that these tasks are completed. As discussed in the section entitled, "IAEA Peer Reviews," above, the United States continues to host OSART missions and followup missions on a regular basis. In addition, U.S. industry representatives continue to participate in OSART missions around the world. Since 2008, INPO has played an important role in the development of the U.S. National Report for the Convention on Nuclear Safety (CNS) and in the peer review process at each review meeting.

IAEA Safety Standards

Through its representation on the IAEA Commission on Safety Standards and the four Safety Standards Committees, the United States is actively participating in IAEA's efforts to review the effectiveness of the international safety standards and to recommend revisions as appropriate. The United States also takes the IAEA safety standards into account in the development of new or revised regulations. The NRC has been a participant in a special working group for the Nuclear Safety Standards Committee tasked with reviewing lessons learned from Fukushima and with incorporating them into IAEA safety standards. This effort is over and above the normal periodic update of IAEA standards. Additional information concerning how the IAEA safety standards inform and guide NRC regulations is discussed in Section 8.1.5.1 of this report.

International Legal Framework

The United States played an active role in the August 2012 Extraordinary Meeting of contracting parties to the CNS; a senior NRC manager served as one of the vice presidents. The Extraordinary Meeting was successful in reaching an agreement upon key revisions to the CNS guidance documents. These revisions immediately strengthened the CNS review process by calling for more robust national reporting and peer reviews. The United States also participated in all meetings of the Working Group on Effectiveness and Transparency, which was created in an effort to evaluate how to further improve the Convention. In addition, the United States participated in the Working Group of Experienced Officers of the CNS and the Joint Convention, which seeks to ensure complementary intent and implementation of these two important international legal instruments. In addition, the NRC has provided two officers to the sixth CNS review meeting to maintain an important tradition of leadership. In this report, the United States has endeavored to address, in detail, all the areas specified in the revised guidance documents and has encouraged other contracting parties to do likewise through its bilateral and multilateral activities. The United States is confident that the changes agreed upon by consensus at the Extraordinary Meeting will lead to more effective and comprehensive reporting and will enable more productive, indepth peer review in the country group meetings. The United States calls on all countries to ratify and fully implement each of the international safety conventions.

Finally, as for civil liability conventions, the United States has ratified the Convention on Supplementary Compensation for Nuclear Damage. The United States has not ratified the Vienna Convention on Civil Liability for Nuclear Damage, and it has not ratified any regional compensation regimes, such as the Paris Convention on Nuclear Third Party Liability. The United States' domestic civil compensation regime is robust and comprehensive. Additional information about the United States' civil compensation regime is discussed in Section 11.1.3 of this report.

Member States Planning To Embark on a Nuclear Power Program

As discussed in Section 8.1.5, the NRC coordinates the International Regulatory Development Partnership collaboratively with Advanced Systems Technology and Management, Inc. The International Regulatory Development Partnership assists countries with emerging nuclear power programs in developing organizational and programmatic resources for regulatory oversight. The program comprises the following three principal elements:

(1) assistance in the development of regulatory agency infrastructure, including organization, staffing, training, and technical support

(2) development of a nuclear regulatory program, including laws, regulations, and regulatory guidance
(3) development and delivery of training for regulatory agency management and staff

The conduct of all work is done in accordance with an action plan that is jointly developed by NRC contractor staff and the management of the participating regulatory agency, based on priorities set by the host country. The program uses guidance from the IAEA and experienced nuclear regulatory programs. Additional information about the program and its accomplishments can be accessed at www.irdp-online.org. The United States also participates actively in IAEA initiatives for emerging countries through a regular budget and extrabudgetary contributions, participation in the development of IAEA documents, and involvement in IAEA peer review services.

Capacity Building

The United States continues to work to ensure the availability of ample resources necessary to ensure a high level of nuclear safety and safe, responsible, and sustainable use of nuclear technologies. The NRC has an active knowledge management program and a program to recruit and develop talented new staff. (Section 8.1.5.2 of this report describes the NRC's knowledge management initiatives and programs.) The NRC also funds a university grant program that provides targeted scholarships and fellowships to promote the study of nuclear-related fields at universities around the United States. Through its Technical Training Center and Professional Development Center, the NRC maintains a comprehensive training program for its own staff and for international personnel on a case-by-case basis as resources permit. The NRC's training experts have also participated in numerous training development activities and workshops at the IAEA. As discussed in the section entitled, "Member States Planning To Embark on a Nuclear Power Program," the NRC is also actively involved in assisting countries that are embarking on new nuclear power programs to obtain necessary training.

Protection of People and the Environment from Ionizing Radiation

As discussed in Section 15 of this report, U.S. radiation control measures are generally founded on radiological risk assessments by the United Nations Scientific Committee on the Effects of Atomic Radiation and the U.S. National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation. The risk management recommendations issued by the International Commission on Radiological Protection and the National Council on Radiation Protection and Measurements reflect these assessments. On the basis of these assessments and recommendations, the U.S. Environmental Protection Agency develops Federal guidance signed by the President of the United States, and "generally applicable radiation standards" for use by the other Federal agencies, including the NRC. U.S. radiation protection programs are based on two principles that are generally consistent with those espoused by the International Commission on Radiological Protection, as follows:

(1) It is known that large doses of ionizing radiation can be deleterious to human health.
(2) It is considered prudent to assume that small doses may also be harmful, with the probability of a deleterious effect being proportional to the dose.

U.S. radiation protection activities apply to the control of radiation exposure to both occupational workers and to members of the public. The NRC is currently exploring the benefits and effects of increasing alignment with new recommendations from the International Commission on Radiological Protection.

Communication and Information Dissemination

As discussed in Section 16.8, the United States places a high priority on effective and transparent communication with the public in the event of an emergency. This communication begins with ensuring that sufficient outreach is conducted so that the public is fully aware of the methods of communication that would be used in such an event.

The United States has participated in each of the IAEA international experts' meetings organized under the auspices of the Action Plan on Nuclear Safety. These meetings have provided beneficial opportunities for U.S technical and policy experts to exchange information and lessons learned with international counterparts who are employing a variety of approaches to address the enhancement of nuclear safety worldwide. These discussions have also helped to enable nuclear regulators, policymakers, and other stakeholders to address the issue of how best to leverage resources to ensure that they are used efficiently while minimizing duplication and overlap. Through its involvement in INPO and the World Association of Nuclear Operators, U.S. industry has also played an important role in promoting enhanced cooperation among nuclear operators and in ensuring that information on enhancing nuclear safety worldwide is broadly shared.

The United States also continues its active participation in activities associated with the International Nuclear and Radiological Event Scale Advisory Committee.

Research and Development

The United States continues to play a lead role in international nuclear safety research. International research is an efficient mechanism for leveraging limited resources and for promoting collaborative work that encourages the use of diverse approaches and viewpoints while discouraging duplication. The United States is working with the Japanese on a bilateral and multilateral basis to address potential post-Fukushima research needs. In particular, the United States has cooperated with Japan to reconstruct the events that led to and resulted from the Fukushima Dai-ichi nuclear accident using the U.S. MELCOR computer code. This cooperative effort relied on event data provided by Japan, and the completion of this study enabled the United States to identify areas in which the MELCOR computer code can or should be improved.

APPENDIX D
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ANNEX 1
U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: U.S. Nuclear Regulatory Commission NUREG-1350, Volume 21, “2009-2010 Information Digest,” August 2009.

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
Arkansas Nuclear One 1 - Entergy Nuclear Operations, Inc.	PWR	2568	12/74	05/34
Arkansas Nuclear One 2 - Entergy Nuclear Operations, Inc.	PWR	3026	03/80	07/38
Beaver Valley 1 - FirstEnergy Nuclear Operating Company	PWR	2900	10/76	01/36
Beaver Valley 2 - FirstEnergy Nuclear Operating Company	PWR	2900	11/87	05/47
Braidwood 1 - Exelon Corp., Exelon Generation Corporation, LLC	PWR	3586.6	07/88	10/26
Braidwood 2 - Exelon Corp., Exelon Generation Corporation, LLC	PWR	3586.6	10/88	12/27
Browns Ferry 1 - Tennessee Valley Authority	BWR	3458	08/74	12/33
Browns Ferry 2 - Tennessee Valley Authority	BWR	3458	03/75	06/34
Browns Ferry 3 - Tennessee Valley Authority	BWR	3458	03/77	07/36
Brunswick 1 - Carolina Power & Light, Co., Progress Energy	BWR	2923	03/77	09/36
Brunswick 2 - Carolina Power & Light, Co., Progress Energy	BWR	2923	11/75	12/34
Byron 1 – Exelon Corp., Exelon Generation Corporation, LLC	PWR	3586.6	09/85	10/24
Byron 2 – Exelon Corp., Exelon Generation Corporation, LLC	PWR	3586.6	08/87	11/26
Callaway – AmerenUE, Union Electric Company	PWR	3565	12/84	10/24

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
Calvert Cliffs 1 - Constellation Energy	PWR	2737	05/75	07/34
Calvert Cliffs 2 - Constellation Energy	PWR	2737	04/77	08/36
Catawba 1 - Duke Energy Carolinas, LLC	PWR	3411	06/85	12/43
Catawba 2 - Duke Energy Carolinas, LLC	PWR	3411	08/86	12/43
Clinton - Exelon Corporation, Exelon Generation Co., LLC	BWR	3473	11/87	09/26
Columbia Generating Station - Energy Northwest	BWR	3486	12/84	12/43
Comanche Peak 1- Luminant Generation Company, LLC	PWR	3612	08/90	02/30
Comanche Peak 2 - Luminant Generation Company, LLC	PWR	3612	08/93	02/33
Cooper - Nebraska Public Power District	BWR	2419	07/74	01/34
Crystal River 3 - Florida Power Corporation, Progress Energy	PWR	2609	03/77	12/16
Davis-Besse - FirstEnergy Nuclear Operating Co.	PWR	2817	07/78	04/17
Diablo Canyon 1 - Pacific Gas & Electric Co.	PWR	3411	05/85	11/24
Diablo Canyon 2 - Pacific Gas & Electric Co.	PWR	3411	03/86	08/25
Donald C. Cook 1 - Indiana/Michigan Power Co.	PWR	3304	08/75	10/34
Donald C. Cook 2 - Indiana/Michigan Power Co.	PWR	3468	07/78	12/37
Dresden 2 - Exelon Corporation, Exelon Generation Co., LLC	BWR	2957	06/70	12/29
Dresden 3 - Exelon Corporation, Exelon Generation Co., LLC	BWR	2957	11/71	01/31
Duane Arnold - FPL Energy Duane Arnold, LLC, Florida Power and Light Co.	BWR	1912	02/75	02/34
Edwin I. Hatch 1 - Southern Nuclear Operating Co.	BWR	2804	12/75	08/34
Edwin I. Hatch 2 - Southern Nuclear Operating Co.	BWR	2804	09/79	06/38

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
Fermi 2 – The Detroit Edison Co.	BWR	3430	01/88	03/25
Fort Calhoun Station – Omaha Public Power District	PWR	1500	09/73	08/33
R.E. Ginna - Constellation Energy	PWR	1775	07/70	09/29
Grand Gulf 1 - Entergy Nuclear Operations, Inc.	BWR	3898	07/85	11/24
H.B. Robinson 2 - Carolina Power & Light Co.	PWR	2339	03/71	07/30
Hope Creek 1 - PSEG Nuclear, LLC	BWR	3840	12/86	04/46
Indian Point 2 - Entergy Nuclear Operations, Inc.	PWR	3216	08/74	09/13
Indian Point 3 - Entergy Nuclear Operations, Inc.	PWR	3216	08/76	12/15
James A. FitzPatrick - Entergy Nuclear Operations, Inc.	BWR	2536	07/75	10/34
Joseph M. Farley 1 - Southern Nuclear Operating Co.	PWR	2775	12/77	06/37
Joseph M. Farley 2 - Southern Nuclear Operating Co.	PWR	2775	07/81	03/41
Kewaunee Power Station - Dominion Energy Kewaune, Inc.	PWR	1772	06/74	12/33
La Salle County 1 - Exelon Corporation, Exelon Generation Co., LLC	BWR	3546	01/84	04/22
La Salle County 2 - Exelon Corporation, Exelon Generation Co., LLC	BWR	3546	10/84	12/23
Limerick 1-Exelon Corporation, Exelon Generation Co., LLC	BWR	3515	02/86	10/24
Limerick 2- Exelon Corporation, Exelon Generation Co., LLC	BWR	3515	01/90	06/29
McGuire 1 - Duke Energy Power Company, LLC	PWR	3411	12/81	06/41
McGuire 2 - Duke Energy Power Company, LLC	PWR	3411	03/84	03/43
Millstone 2 – Dominion Nuclear Connecticut, Inc., Dominion Generation	PWR	2700	12/75	07/35

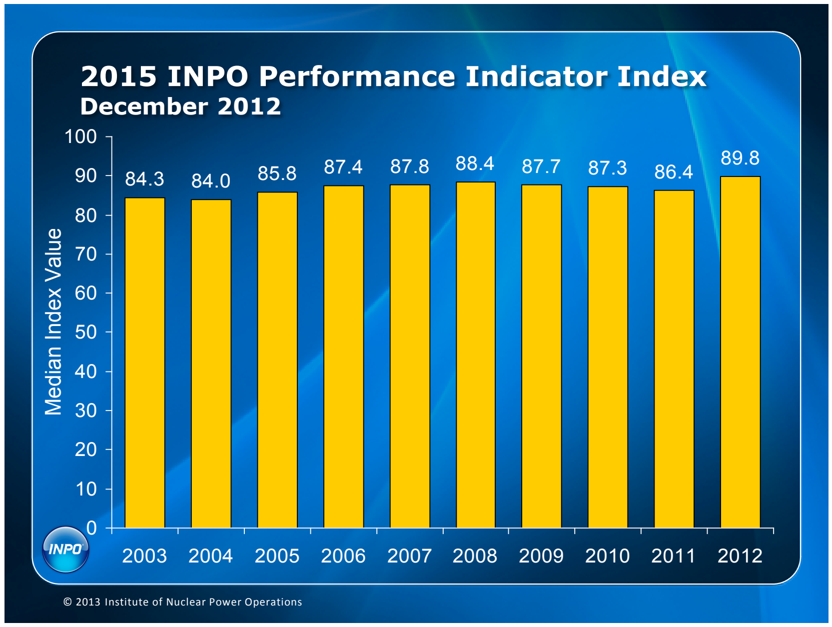
Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWth)	Operating Lifetime	
Millstone 3 - Dominion Nuclear Connecticut, Inc., Dominion Generation	PWR	3650	04/86	11/45
Monticello - Nuclear Management Co.	BWR	1775	06/71	09/30
Nine Mile Point 1 - Constellation Energy	BWR	1850	12/69	08/29
Nine Mile Point 2 - Constellation Energy	BWR	3988	03/88	10/46
North Anna 1 Virginia Electric & Power Co., Dominion Generation	PWR	2940	06/78	04/38
North Anna 2 - Virginia Electric & Power Co., Dominion Generation	PWR	2940	12/80	08/40
Oconee 1 - Duke Energy Power Company, LLC	PWR	2568	07/73	02/33
Oconee 2 - Duke Energy Power Company, LLC	PWR	2568	09/74	10/33
Oconee 3 - Duke Energy Power Company, LLC	PWR	2568	12/74	12/34
Oyster Creek - AmerGen Energy Co., LLC, Exelon Corporation	BWR	1930	12/69	04/29
Palisades - Entergy Nuclear Operations, Inc.	PWR	2565	12/71	03/31
Palo Verde 1 - Arizona Public Service Company	PWR	3990	01/86	06/45
Palo Verde 2 - Arizona Public Service Company	PWR	3990	09/86	04/46
Palo Verde 3 - Arizona Public Service Company	PWR	3990	01/88	11/47
Peach Bottom 2 Exelon Corp., Exelon Generation Corporation, LLC	BWR	3514	07/74	08/33
Peach Bottom 3 Exelon Corp., Exelon Generation Corporation, LLC	BWR	3514	12/74	07/34
Perry 1 - FirstEnergy Nuclear Operating Co.	BWR	3758	11/87	03/26
Pilgrim 1 - Entergy Nuclear Operations, Inc.	BWR	2028	12/72	06/32
Point Beach 1 - FLP Energy Point Beach, LLC, Florida Power and Light Co.	PWR	1800	12/70	10/30
Point Beach 2 - FLP Energy Point Beach, LLC, Florida Power and Light Co.	PWR	1800	10/72	03/33
Prairie Island 1 - Nuclear Management Co.	PWR	1677	12/73	08/33

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWth)	Operating Lifetime	
Prairie Island 2 - Nuclear Management Co.	PWR	1677	12/74	10/34
Quad Cities 1 Exelon Corporation, Exelon Generation Co., LLC	BWR	2957	02/73	12/32
Quad Cities 2 - Exelon Corporation, Exelon Generation Co., LLC	BWR	2957	03/73	12/32
River Bend 1 - Entergy Nuclear Operations, Inc.	BWR	3091	06/86	08/25
Salem 1 - PSEG Nuclear, LLC	PWR	3459	06/77	08/36
Salem 2 - PSEG Nuclear, LLC	PWR	3459	10/81	04/40
San Onofre 2 - Southern California Edison Co.	PWR	3438	08/83	02/22
San Onofre 3 - Southern California Edison Co.	PWR	3438	04/84	11/22
Seabrook 1 - FPL Energy Seabrook, LLC	PWR	3648	08/90	03/30
Sequoyah 1 - Tennessee Valley Authority	PWR	3455	07/81	09/20
Sequoyah 2 - Tennessee Valley Authority	PWR	3455	06/82	09/21
Shearon Harris 1 - Carolina Power & Light Co.	PWR	2900	05/87	10/46
South Texas Project 1 - STP Nuclear Operating Co.	PWR	3853	08/88	08/27
South Texas Project 2 - STP Nuclear Operating Co.	PWR	3853	06/89	12/28
St. Lucie 1 - Florida Power & Light Co.	PWR	2700	12/76	03/36
St. Lucie 2 - Florida Power & Light Co.	PWR	2700	08/83	04/43
Surry 1 - Dominion Generation	PWR	2857	12/72	05/32
Surry 2 - Dominion Generation	PWR	2857	05/73	01/33
Susquehanna 1 - PPL Susquehanna, LLC	BWR	3952	06/83	07/42
Susquehanna 2 - PPL Susquehanna, LLC	BWR	3952	02/85	03/44
Three Mile Island 1 - AmerGen Energy Co., LLC	PWR	2568	09/74	04/34
Turkey Point 3 - Florida Power & Light Co.	PWR	2300	12/72	07/32

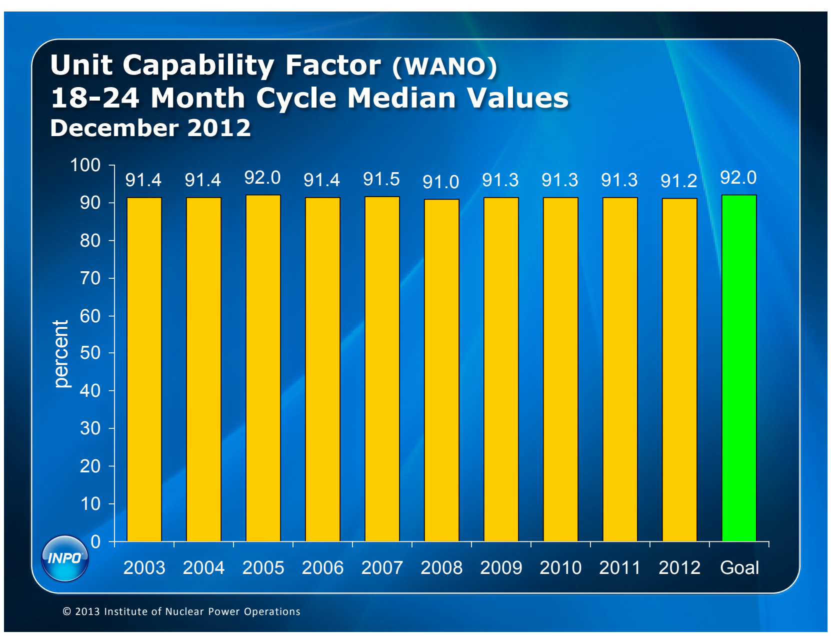
Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWth)	Operating Lifetime
Turkey Point 4 - Florida Power & Light Co.	PWR	2300	09/73 04/33
V.C. Summer - South Carolina Electric & Gas Co.	PWR	2900	01/84 08/42
Vermont Yankee - Entergy Nuclear Operations, Inc.	BWR	1912	11/72 03/32
Vogtle 1 - Southern Nuclear Operating Co.	PWR	3625	06/87 01/47
Vogtle 2 - Southern Nuclear Operating Co.	PWR	3625	05/89 02/49
Waterford 3 - Entergy Nuclear Operations, Inc	PWR	3716	09/85 12/24
Watts Bar 1 - Tennessee Valley Authority	PWR	3459	05/96 11/35
Wolf Creek 1 - Wolf Creek Nuclear Operating Corporation	PWR	3565	09/85 03/45

ANNEX 2
U.S. NUCLEAR ELECTRIC INDUSTRY PERFORMANCE INDICATOR GRAPHS

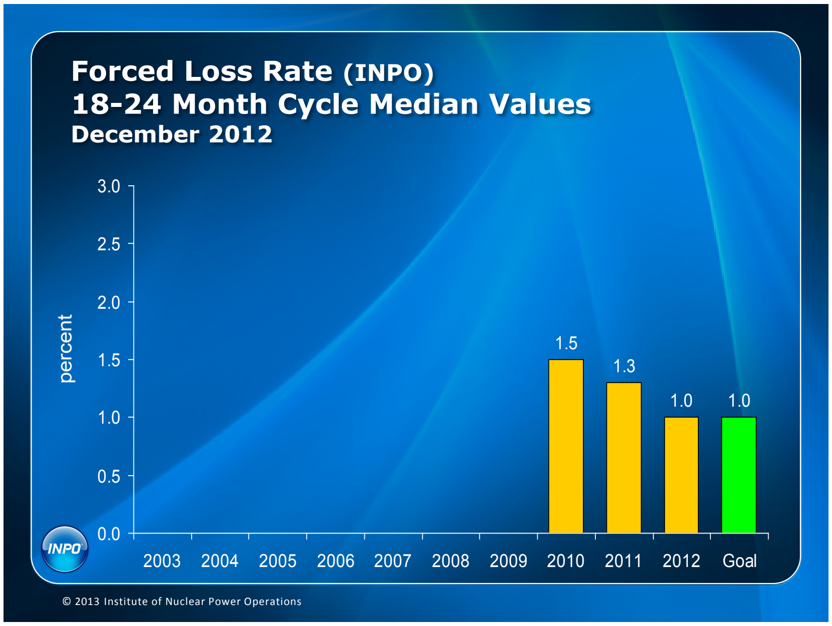
Performance Indicator Index
December 2012



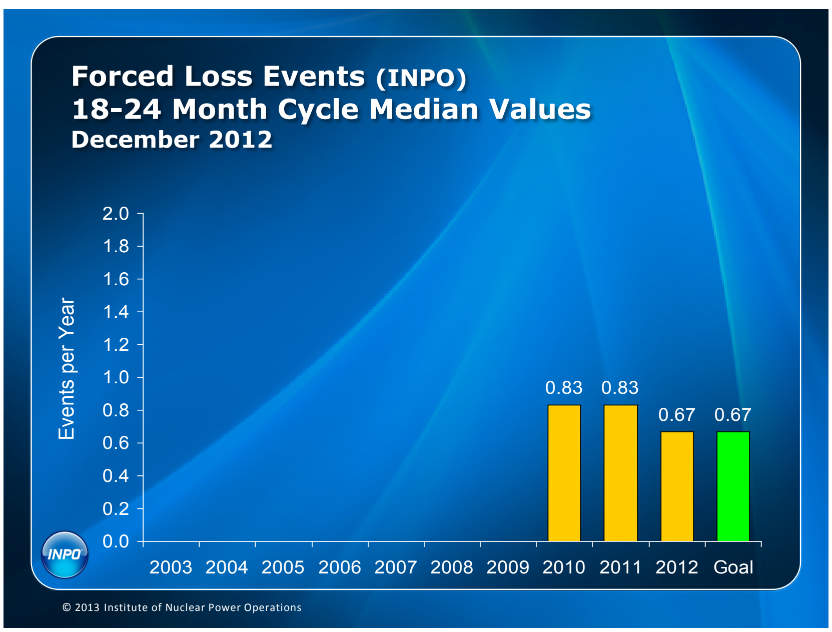
Unit Capability Factor
Median Values
December 2012



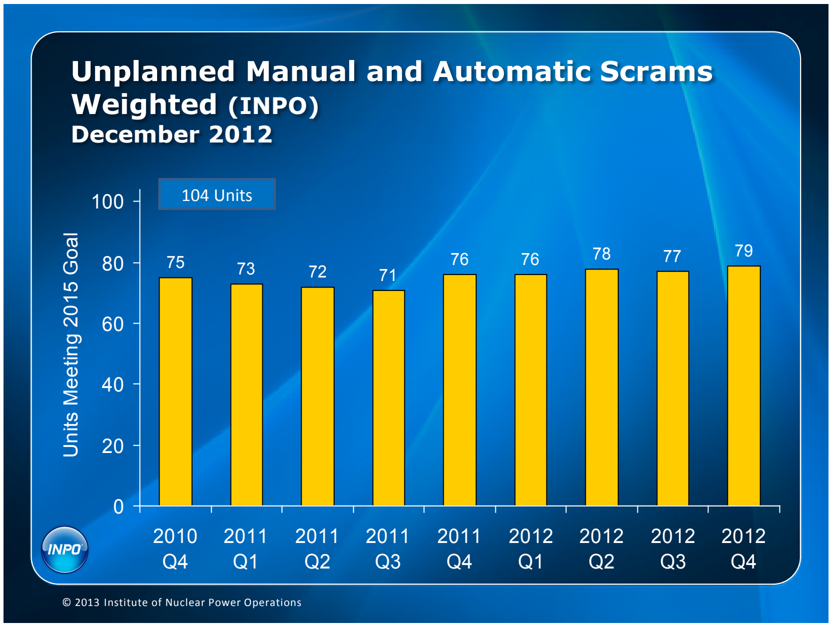
Forced Loss Rate
Median Values
December 2012



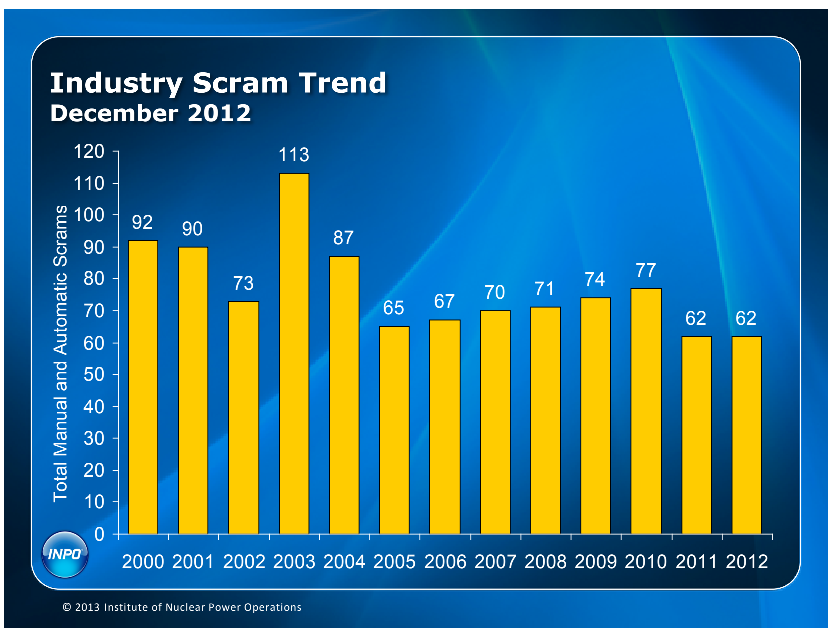
Forced Loss Events
Median Values
December 2012



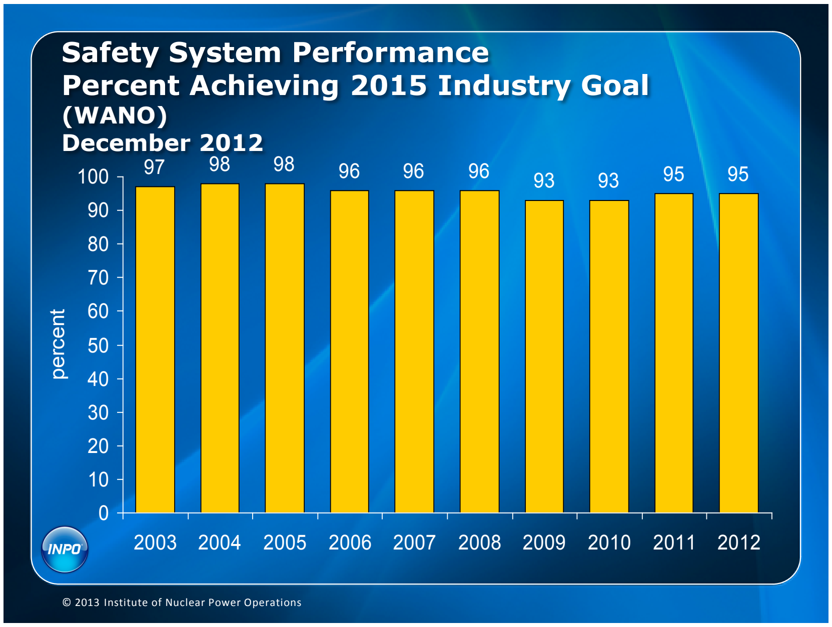
Unplanned Manual and Automatic Scrams
Weighted
December 2012



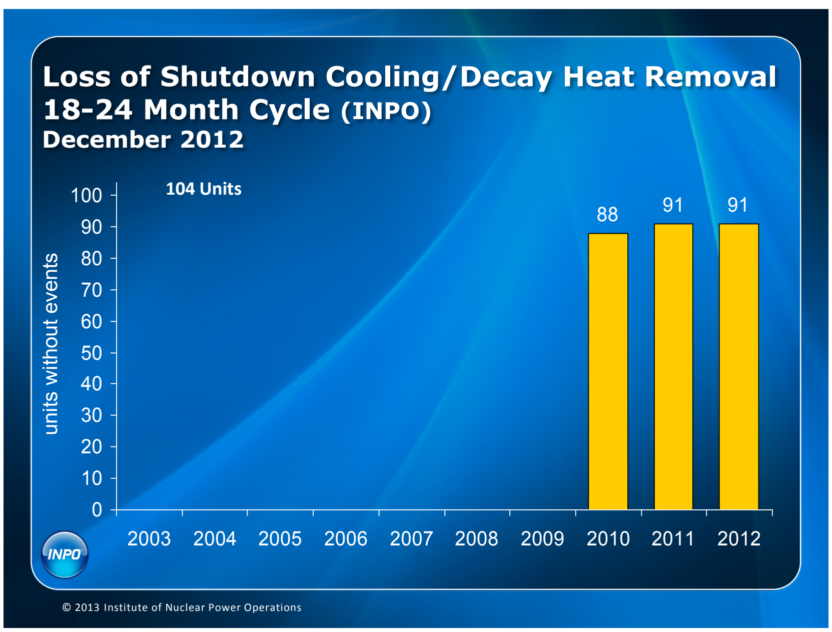
Industry Scram Trend
December 2012



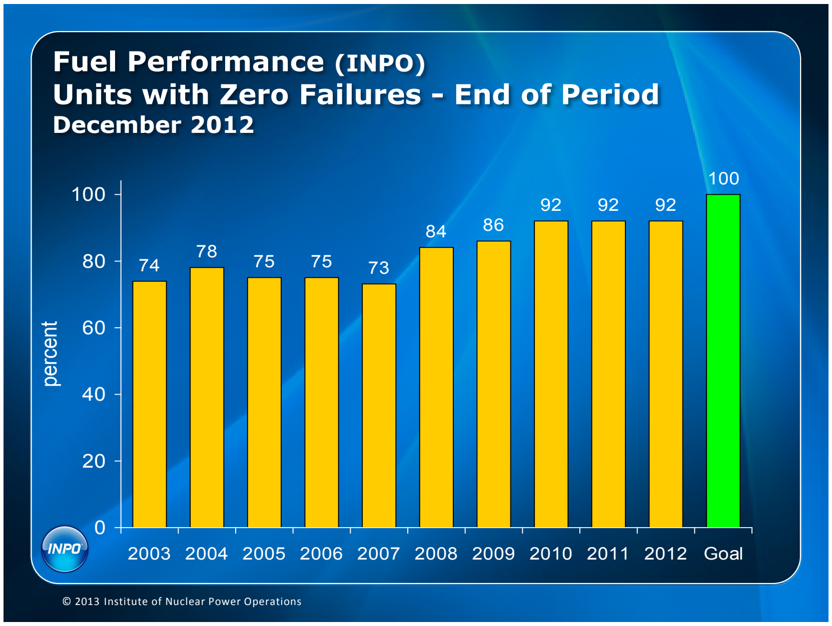
Safety System Performance
December 2012



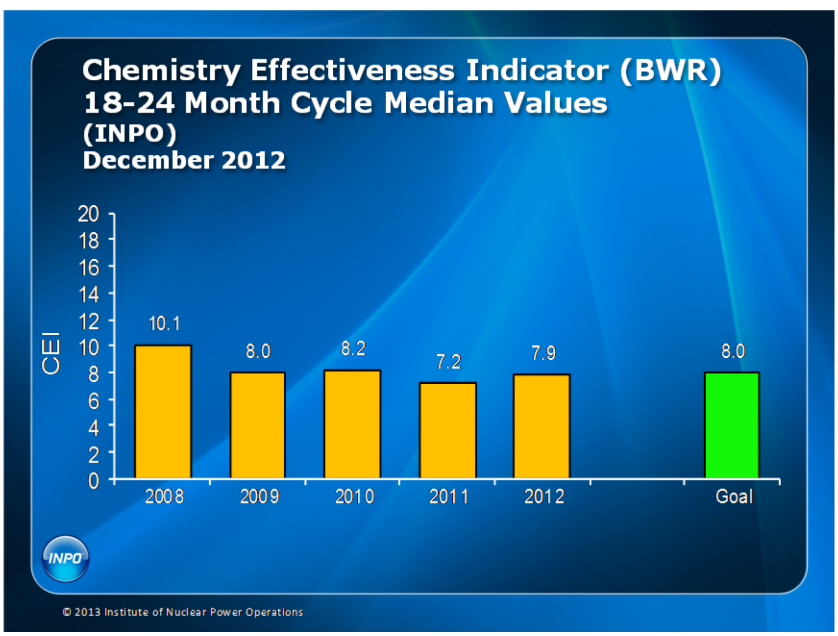
Loss of Shutdown Cooling / Decay Heat Removal
December 2012



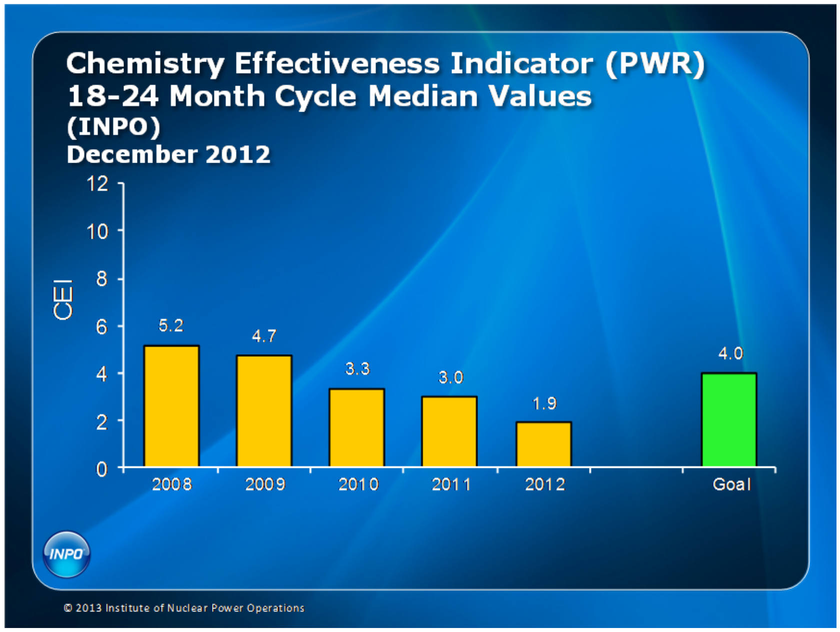
Fuel Performance
December 2012



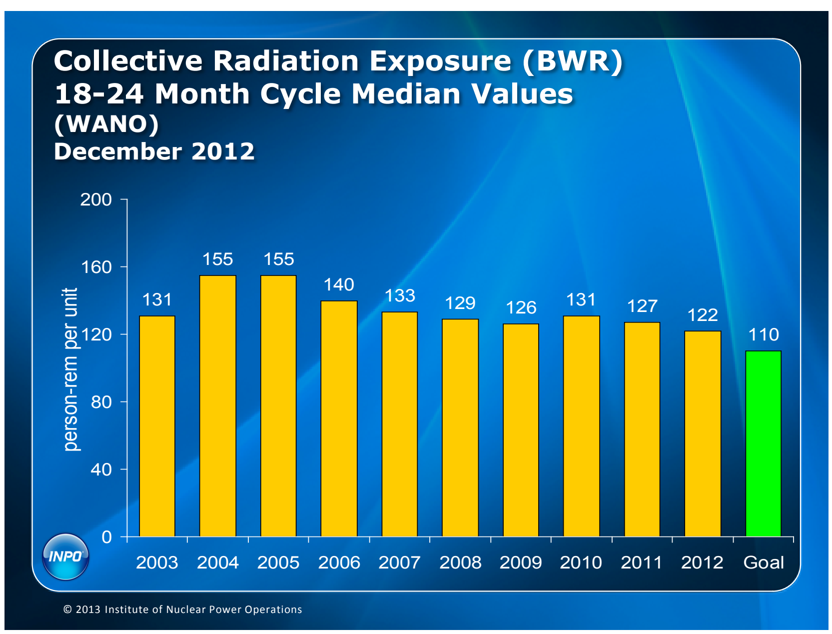
Chemistry Effectiveness Indicator (BWR)
Median Values
December 2012



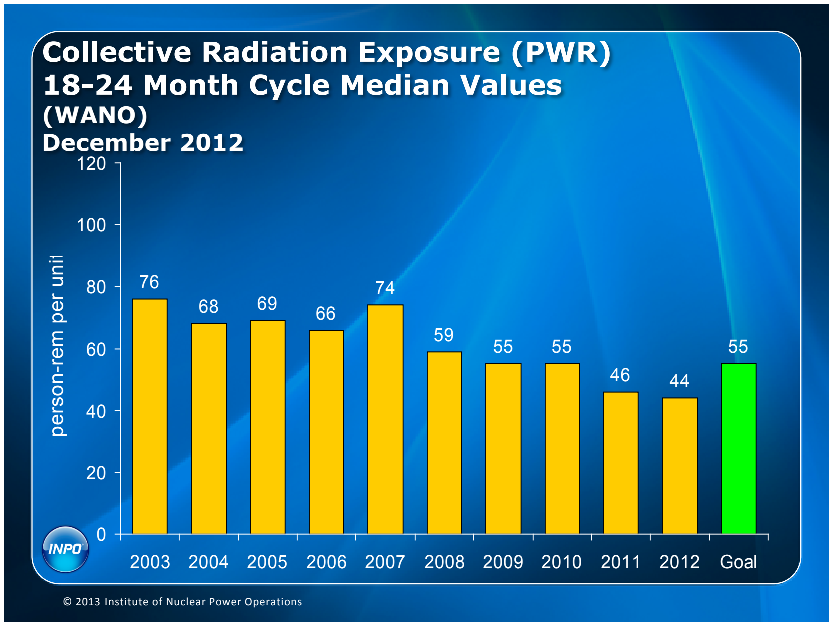
Chemistry Effectiveness Indicator (PWR)
Median Values
December 2012



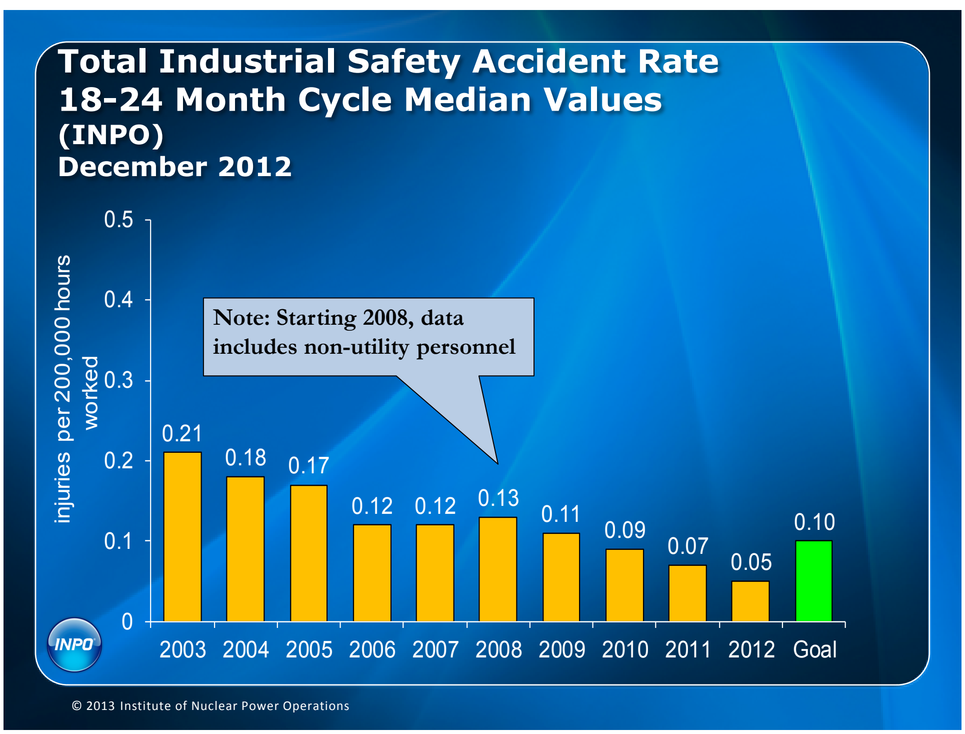
Collective Radiation Exposure (BWR)
Median Values
December 2012



Collective Radiation Exposure (PWR)
Median Values
December 2012



Total Industrial Safety Accident Rate
Median Values
December 2012



NRC FORM 335 (12-2010) NRCMD 3.7

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Washington, D.C. 20555-0001

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10. SUPPLEMENTARY NOTES

This report is an update to NUREG-1650, Revision 4

11. ABSTRACT (200 words or less)

The United States (U.S.) Nuclear Regulatory Commission (NRC) has updated NUREG-1650, Revision 4, "The United States of America Fifth National Report for the Convention on Nuclear Safety," published in September 2010, and will submit this report for peer review at the sixth review meeting of the Convention on Nuclear Safety at the International Atomic Energy Agency in Vienna Austria, in March-April 2014. This report addresses the safety of land-based commercial nuclear power plants in the U.S. It demonstrates how the U.S. Government achieves and maintains a high level of nuclear safety worldwide by enhancing national measures and international cooperation, and by meeting the obligations of all the articles established by the Convention. These articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, the priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation. Similar to the U.S. National Reports issued in 2007 and 2010, this document includes a section developed by the Institute of Nuclear Power Operations (INPO) describing work done by the U.S. nuclear industry to ensure safety. The prime responsibility for the safety of a nuclear installation rests with the license holder; therefore, Part 3 explains how the nuclear industry maintains and improves nuclear safety.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Convention on Nuclear Safety, CNS, treaties, nuclear plants, programs, challenges, reviews, legislation, regulation, reactor oversight, public participation, probabilistic risk analysis, PRA, performance-based, risk-based, quality, safety, knowledge management, human capital, security, siting, design, construction, operations, new reactors, radiation protection, emergency preparedness, financial, human factors, Institute of Nuclear Power Operations, INPO, export, import, peer review

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