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NUREG-1650
Revision 6

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

Office of Nuclear Reactor Regulation

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NUREG-1650
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The United States of America Seventh National Report for the Convention on Nuclear Safety

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Prepared by
U.S. Nuclear Regulatory Commission (NRC)
Institute of Nuclear Power Operations (INPO)

Office of Nuclear Reactor Regulation

ABSTRACT

The U.S. Nuclear Regulatory Commission has prepared Revision 6 to NUREG-1650, “The United States of America Seventh National Report for the Convention on Nuclear Safety,” for submission for peer review at the seventh review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2017. 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 also addresses the principles of the Vienna Declaration adopted by the Contracting Parties in February 2015.

Similar to the U.S. National Report issued in 2013, 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 6 to NUREG-1650, “The United States of America Seventh National Report for the Convention on Nuclear Safety,” for submission for peer review at the seventh review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2017. 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 issues identified through the peer review conducted during the sixth review meeting in April 2014 and discusses challenges and issues that have arisen since that time. The sixth review meeting identified the following six U.S. challenges:

(1) Fukushima-related activities
(2) risk-informed fire protection regulations requiring extensive resources for the transition to National Fire Protection Association 805
(3) ensuring continuity during the oversight transition from plant construction to operation
(4) nuclear industry strategy
(5) reporting status of the periodic safety review gap being developed
(6) status of the NRC’s work on the issuance of possible license renewals to operate beyond 60 years (i.e., subsequent license renewal)

This report discusses the status of safety issues raised in the sixth U.S. National Report, including implementation of Fukushima lessons, nondestructive examinations, concrete degradation, cumulative effects of regulations, economic consequences, counterfeit and fraudulent items, and construction inspection. The report also addresses the following safety and regulatory issues that have necessitated significant attention since 2013:

(1) baffle-former bolts
(2) digital instrumentation and control systems
(3) open phase conditions in electric power systems
(4) risk-informing regulations and processes
(5) spent fuel pool neutron-absorbing materials
(6) staff readiness to transition plants from construction to operations
(7) staff readiness to transition plants from operation to decommissioning
(8) subsequent license renewal
(9) implementation of the principles of the Vienna Declaration on Nuclear Safety
(10) Project Aim

The Institute of Nuclear Power Operations 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, including challenges and the implementation of lessons learned from the Fukushima accident.

ACKNOWLEDGMENTS

Contributors to this report include the following technical and regulatory experts from the U.S. Nuclear Regulatory Commission:

Akstulewicz, Frank
Algama, Don
Allwein, Russell
Anderson, James
Anderson, Joseph
Armstrong, Kenneth
Arndt, Steven
Bales, Michelle
Barczy, Theresa
Basu, Sudhamay
Bauer, Laurel
Beltz, Terry
Benowitz, Howard
Billings, Sally
Billoch, Araceli
Biro, Mihaela
Bloom, Steve
Bowman, Gregory
Boyce, Tom
Buford, Angela
Burke, John
Campbell, Steve
Campbell, Tison
Carte, Norbet
Cauffman, Chris
Chen, Yen-Ming
Clark, Brooke
Clark, Jefferson
Clark, Sheldon
Coyne, Kevin
Cruz, Holly
Cruz, Zahira
Culp, Lisa-Anne
Cumblidge, Stephen
Cushing, Jack
Decker, David
Dhir, Neha
Dickson, Elijah
Dinsmore, Steve
Dougherty, Jay
Doyle, Dan
Ennis, Rick
Ferkile, Andrea
Ferrell, Kimberly
Flanders, Scott
Foggie, Kirk
Folk, Kevin
Fong, CJ
Frahm, Ron
Franovich, Rani
Freeman, Eric
Frumkin, Dan
Gallucci, Ray
Gardocki, Stan
Garry, Steve
Gaslevic, Jim
Gavrilas, Mirela
Gendelman, Adam
Gennardo, David
Gibson, Lauren
Goetz, Robert
Gordon, Dennis
Grant, Jeffrey
Green, Kimberly
Hardies, Bob
Hart, Michelle
Hayes, Michelle
Held, Wesley
Henderson, Karen
Hill, Brittan
Hiser, Allen
Hollcraft, Zachary
Hopkins, Jon
Huffert, Antony
Huffman, Bill
Imboden, Stacey
Iyengar, Raj
Jackson, Chris
Jackson, Deborah
Jarriel, Lisa
Johnson, Don
Kahler, Robert
Karwoski, Kenneth
Keefe, Molly
Keim, Andrea
Kichline, Michelle
Kim, Grace
Klein, Alex
Koenick, Stephen
Kohen, Marshall
Kolb, Tim
Kozak, Tom
Kratchman, Jessica
Kugler, Andrew
Kulesa, Gloria
Kuntz, Robert
Kuritzky, Alan
Lane, John
Lappert, Glenna
Laur, Steve
Lee, Eric
Lee, Richard
Lehman, Bryce
Lewis, Doris
Lian, Jocelyn
Lising, Jason
Lois, Kosmas
Lukes, Robert
Lupold, Tim
Lyons, Sara
Lyons-Burke, Kathy
Mahlahla, LaTonya
Martin, Kamishan
Mathew, Roy
McCartin, Timothy
McIntyre, David
McKirgan, John
Merzke, Daniel
Miller, Barry
Miller, Chris
Miller, Keith
Milligan, Patricia
Mizuno, Geary
Monninger, John
Montecalvo, Michael
Mossman, Timothy
Moyer, Carol
Muller, David
Murray, Charles
Nakoski, John
Nourbakhsh, Hossein
Oesterle, Eric

Olmstead, Joan
Orders, William
Orr, Mark
Quinones, Lauren
Palmrose, Donald
Pantalo, Charity
Pasquale, Dan
Patel, Ami
Pedersen, Renee
Pedersen, Roger
Peduzzi, Francis
Peralta, Juan
Portillo, Veronica
Prescott, Paul
Prescott, Peter
Purnell, Blake
Rautzen, William
Reed, Timothy
Rini, Brett
Rivera, Alison
Robinson, Jay
Rodriguez, Veronica
Rosales-Cooper, Cindy
Rosenberg, Stacey
Ross-Lee, Mary-Jane
Rossi, Mathew
Rothschild, Trip
Ruffin, Steve
Sanders, Serita
Schofer, Fred
Schrader, Eric
Scott, Cathy
Seshagiri Rao, Tammara
Shoop, Undine
Sieracki, Diane
Sigmon, Rebecca
Smith, Shawn
Smith, Will
Syndor, Russell
Tadesse, Rebecca
Tallarico, Alison
Taylor, Robert
Thomas, George
Thompson, Catherine
Truong, Tung
Telson, Ross
Temple, Jeffrey
Thompson, Catherine
Trocine, Leigh
Vanden Berghe, John
Vechioli, Lucieann
Watson, Bruce
Waugh, Andrew
Weber, Carl
Weerakkody, Sunil
Westreich, Barry
Whited, Jeffrey
Whitman, Jennifer
Wilkins, Lynnea
Williams, Donna
Williams, Kevin
Wong, Albert
Wong, Emma
Wood, Kent
Wray, John
Yip, Brian
Yoder, Matt

Contributors to this report include the following experts from the Institute of Nuclear Power Operations:

Aitken, Steve
Berko, Dave
Cox, Tammy
Crane, Randy
Crawley, Carmen
Gambrill, Bob
Hembree, Dave
Johnson, Steve
Klodnicki, Mike
Koves, Ken
Love, Tammy
Martinez, Ramon
Masters, Glen
Nichols, Steve
Place, Jeff
Simon, Shawn
Spinnato, Roger
Weaver, Claire

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
APR	advanced power reactor
ASME	American Society of Mechanical Engineers
BSAF	Benchmark Study of the Accident at the Fukushima Dai-ichi
BRIIE	baseline risk index for initiating events
BWR	boiling-water reactor
CEO	chief executive officer
CFR	*Code of Federal Regulations*
CFSI	Counterfeit, Fraudulent, Suspect Items
CNS	Convention on Nuclear Safety
ConE	construction experience program
DG	draft regulatory guide
DHS	U.S. Department of Homeland Security
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
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
FR	*Federal Register*
FY	fiscal year
GL	generic letter
IAEA	International Atomic Energy Agency
ICES	INPO Consolidated Event System
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IER	INPO event reports
IN	information notice
INPO	Institute of Nuclear Power Operations
IP	inspection procedure
IPSR	INPO performance summary report
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
LOCA	loss of coolant accident
LOOP	loss of offsite power
MD	management directive
MWe	megawatt electric
MWt	megawatt thermal
NANTeL	National Academy for Nuclear Training e-Learning
NATF	North American Transmission Forum
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NEPA	National Environmental Policy Act
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
PML	performance monitoring leaders
POC	performance objectives and criteria
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RFI	request for information
RG	regulatory guide
RIS	regulatory issue summary
SAMGs	severe accident management guidelines
SAREF	SAfety REsearch Opportunities post-Fukushima
SAT	systems approach to training
SBO	station blackout
SEP	Systematic Evaluation Program
SFP	spent fuel pool
SOER	Significant Operating Experience Report
SSC	structure, system, and component
TI	temporary instruction
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

PART 1

INTRODUCTION

This section describes the purpose and structure of the "United States of America Seventh National Report for the Convention on Nuclear Safety," and provides a summary of changes in the seventh 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 seventh 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 seventh U.S. National Report, principally using agency information that is publicly available. This updated report follows the format of the sixth U.S. National Report published in 2013 and is designed to be a standalone document. Therefore, this report duplicates some of the information presented in the 2013 report. To facilitate peer review, Table 1 includes a summary of the main changes to the report. This table in Part 1 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.

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

Similar to the 2013 report, Part 3 of this document includes a contribution by the Institute of 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, the agency's Strategic Plan, references, and a list of nuclear plants in the United States.

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 sixth report.

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

Report Section		Change
Abstract		Updated to add discussion about the sixth CNS and the Vienna Declaration on Nuclear Safety
Executive Summary		Updated to add discussion about the sixth CNS and the Vienna Declaration on Nuclear Safety
PART 1		
Introduction		Updated to add discussion about the sixth CNS
Purpose and Structure of This Report		Updated to add discussion about the sixth CNS
Summary of Changes to the Sixth U.S. National Report		Updated table
Section 1	SUMMARY	Updated to add discussion about the sixth CNS
1.1	The U.S. Policy toward Nuclear Activities	No changes
1.1.1	Regulatory Body Organizational Values	No changes
1.1.2	Regulatory Body Challenges	Updated to add discussion on most recent NRC Strategic Plan and Inspector General report
1.2	National Nuclear Programs	Editorial changes only
1.2.1	Reactor Oversight Process	Updated to add discussion about the self-assessment
1.2.2	License Renewal	Updated to add discussion about units entering the 41st year of operation
1.2.3	Power Uprate Program	No changes

Report Section		Change
1.2.4	New Reactor Licensing	Summarized. Updated to add discussion about applications under review.
1.3	Safety and Regulatory Issues, and Regulatory Accomplishments	Editorial changes only
1.3.1	Safety and Regulatory Issues Discussed in the Sixth U.S. National Report	Completely updated to add current status
1.3.2	Current Safety and Regulatory Issues	Completely updated. Discusses 7 new topics.
1.3.3	Major Regulatory Accomplishments	Completely updated. Discusses 6 new topics.
1.4	International Peer Reviews and Missions	Editorial changes only
1.4.1	Convention on Nuclear Safety	Updated to include results from sixth CNS report and Rapporteurs' findings. A new subsection on the Vienna Declaration on Nuclear Safety has been added.
1.4.2	Integrated Regulatory Review Service	Updated to include reference to mission results
1.4.3	Operational Safety Review Team	Updated to include reference to mission results
PART 2		
Article 6	EXISTING NUCLEAR INSTALLATIONS	Updated to state that the article addresses the Vienna Declaration on Nuclear Safety
6.1	Introduction	Updated to add performance goals
6.2	Nuclear Installations in the United States	Updated to include status of plants in operation and shutdowns
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 discuss current plant performance status
6.3.3	Industry Trends Program	Updated to discuss 2014 trends results
6.3.4	Accident Sequence Precursor Program	Updated to include a discussion about the accident sequence precursor program status report issued in 2015
6.3.5	Operating Experience Program	Editorial changes only
6.3.6	Generic Issues Program	Updated to reflect changes to streamline the program
6.3.7	Rulemaking	No changes
6.3.8	Fire Protection Regulation Program	Updated to discuss challenges and updates on the transition to risk-informed rule
6.3.9	Decommissioning	Updated to discuss new decommissioning rulemaking
6.3.10	Reactor Safety Research Program	No changes
6.3.11	Public Participation	Updated to refine discussion on rulemaking process
6.4	Lessons Learned from Fukushima	Editorial changes only
6.5	Vienna Declaration on Nuclear Safety	New section
Article 7	LEGISLATIVE AND REGULATORY FRAMEWORK	Updated to state that no changes to U.S. legislative framework were made post-Fukushima
7.1	Legislative and Regulatory Framework	Updated to add discussion on the Convention on the Physical Protection on Nuclear Material
7.2	Provisions of the Legislative and Regulatory Framework	No changes

Report Section		Change
7.2.1	National Safety Requirements and Regulations	Updated to refine discussion on rulemaking process
7.2.2	Licensing of Nuclear Installations	Editorial changes only
7.2.3	Inspection and Assessment	Editorial changes only
7.2.4	Enforcement	Editorial changes only
7.3	Lessons Learned from Fukushima	Section deleted. No changes to U.S. legislative framework post-Fukushima.
Article 8	REGULATORY BODY	Updated to state that regulatory actions post-Fukushima are discussed in Section 1
8.1	The Regulatory Body	No changes
8.1.1	Mandate	No changes
8.1.2	Authority and Responsibilities	No changes
8.1.2.1	Scope of Authority	No changes
8.1.2.2	The NRC as an Independent Regulatory Agency	Editorial changes only
8.1.3	Structure of the Regulatory Body	No changes
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	Updated to reflect organizational changes
8.1.3.4	Advisory Committees	No changes
8.1.3.5	Atomic Safety and Licensing Board Panel	No changes
8.1.3.6	Office of the Inspector General	Editorial changes only
8.1.4	Position of the NRC in the Governmental Structure	No changes
8.1.4.1	Executive Branch	Updated to discuss role of the National Security Council
8.1.4.2	The States (i.e., of the United States)	Editorial changes only
8.1.4.3	Congress	No changes
8.1.5	International Responsibilities and Activities	Updated throughout
8.1.5.1	International Standards	Updated to include committee representation and efforts to harmonize with NRC guidance
8.1.5.2	Integrated Regulatory Review Service Mission	Updated to add discussion on findings and followup mission results
8.1.5.3	Operational Safety Assessment Review Teams	Updated to add discussion on findings and upcoming mission
8.1.6	Financial and Human Resources	No changes
8.1.6.1	Financial Resources	Updated to add funds for fiscal year 2015
8.1.6.2	Human Resources	Updated to discuss survey findings and knowledge management initiatives
8.1.7	Openness and Transparency	Updated to include most recent numbers associated to public outreach activities
8.2	Separation of Functions of the Regulatory Body from Those of Bodies Promoting Nuclear Energy	Updated to refine discussion on means by which effective separation and independence is ensured
8.3	Fukushima Lessons Learned	Deleted section. Regulatory actions post-Fukushima are discussed in Section 1.
8.3	Ethics Rules Applying to NRC Employees and Former Employees	New section. Replaces the original 8.3 section.

Report Section		Change
Article 9	RESPONSIBILITY OF THE LICENSE HOLDER	Updated to state that overall responsibility of the license holder did not change post-Fukushima
9.1	Introduction	No changes
9.2	The Licensee's Primary Responsibility for Safety	Updated to refine discussion on means to ensure licensee has resources for managing an accident
9.3	The NRC Enforcement Program	Updated to discuss recent enforcement actions
9.4	Openness and Transparency	Editorial changes only
9.5	Fukushima Lessons Learned	Deleted section. Overall responsibility of the license holder did not change post-Fukushima.
Article 10	PRIORITY TO SAFETY	Updated to state that no changes to policies on priority to safety have been made as a result of Fukushima.
10.1	Background	Updated throughout. New references.
10.2	Probabilistic Risk Assessment Policy	No changes
10.3	Applications of Probabilistic Risk Assessment	Editorial changes only
10.3.1	Risk-Informed Special Treatment	Updated to discuss update on 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	Updated to discuss updates to standard
10.3.5	Level 3 Probabilistic Risk Assessment Project	New section
10.4	Safety Culture	Restructured to include the policy statement discussion in Section 10.4.1
10.4.1	Safety Culture Policy Statement	Renamed. Updated to discuss new safety culture traits.
10.4.2	The NRC Monitoring of Licensee Safety Culture	Renumbered
10.4.2.1	Background	Renumbered. No wording changes.
10.4.2.2	Enhanced Reactor Oversight Process	Renumbered. Updated to discuss issuance of revised procedures.
10.4.3	The NRC Safety Culture	Renumbered. Updated throughout. Discusses safety culture components.
10.5	Managing the Safety and Security Interface	Updated to add new references
10.6	Fukushima Lessons Learned	Section deleted. No changes to policies on priority to safety as a result of Fukushima.
Article 11	FINANCIAL AND HUMAN RESOURCES	Updated to state that no changes in financial resources measures have taken place post-Fukushima
11.1	Financial Resources	Updated to discuss financial qualification regulations
11.1.1	Financial Qualifications for Construction and Operations	Minor editorial change in the section name
11.1.1.1	Construction Permit Reviews	Editorial changes only
11.1.1.2	Operating License Reviews	Editorial changes only
11.1.1.3	Combined License Application Reviews	Updated to refine discussion on application requirements
11.1.1.4	Postoperating License Nontransfer Reviews	Updated to discuss issuance of interim staff guidance

Report Section		Change
11.1.1.5	Reviews of License Transfers	Updated to discuss financial qualification regulations
11.1.2	Financial Qualifications Program for Decommissioning	Updated to discuss decommissioning funding regulations
11.1.3	Financial Protection Program for Liability Claims Arising from Incidents	Minor editorial change in the section name. Updated the retrospective premium pool requirements
11.1.4	Insurance Program for Onsite Property Damages Arising from Accidents	Editorial changes only
11.2	Regulatory Requirements for Qualifying, Training, and Retraining Personnel	No changes
11.2.1	Governing Documents and Process	Updated references
11.2.2	Experience	No changes
11.3	Fukushima Lessons Learned	Section deleted. No changes to licensee financial requirements have taken place post-Fukushima.
Article 12	HUMAN FACTORS	Editorial changes only
12.1	Goals and Mission of the Program	No changes
12.2	Program Elements	No changes
12.3	Significant Regulatory Activities	Editorial changes only
12.3.1	Human Factors Engineering	Approved extended power uprates and editorial changes. Minor change in section title.
12.3.2	Emergency Operating Procedures and Plant Procedures	Updated status on Fukushima activities and information on mitigating strategies rule.
12.3.3	Shift Staffing	Updated to include discussion on staffing plan validation
12.3.4	Fitness for Duty	Updated reference on fatigue and fitness-for-duty
12.3.5	Human Factors Information System	Update on database
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	Update on status of Fukushima activities
Article 13	QUALITY ASSURANCE	Updated to state that regulatory actions post-Fukushima are discussed in Section 1
13.1	Background	No changes
13.2	Regulatory Policy and Requirements	Editorial changes only
13.2.1	Appendix A to 10 CFR Part 50	No changes
13.2.2	Appendix B to 10 CFR Part 50	No changes
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	Reference update
13.3.2	Guidance for Design and Construction Activities	No changes
13.3.3	Guidance for Operational Activities	Reference update
13.4	Quality Assurance Programs	Reference update
13.5	Quality Assurance Audits Performed by Licensees	No changes
13.5.1	Audits of Vendors and Suppliers	Editorial changes only
13.6	Fukushima Lessons Learned	Section deleted. Fukushima actions are discussed in Section 1.
13.6	Vendor Inspection Program	New section. Replaces the original 13.6 section.

Report Section		Change
Article 14	ASSESSMENT AND VERIFICATION OF SAFETY	Editorial changes only
14.1	Ensuring Safety Assessments throughout Plant Life	Editorial changes only
14.1.1	Assessment of Safety	Updated to include information on plant licensing basis.
14.1.2	Maintaining the Licensing Basis	Editorial changes only
14.1.2.1	Governing Documents and Process	No changes
14.1.1.2	Regulatory Framework for the Restart of Browns Ferry, Unit 1	No changes
14.1.3	Power Uprates	No changes
14.1.3.1	Governing Documents and Process	Editorial changes only
14.1.3.2	Experience	Updated information on power uprates to date, including details on Peach Bottom and Monticello
14.1.4	License Renewal	No changes
14.1.4.1	Governing Documents and Process	Updated discussion of the continued storage rule, and added new references and information on new guidance documents
14.1.4.2	Experience	Updated discussion about renewed license to date
14.1.4.3	Operating Beyond 60 Years	Completely updated
14.1.5	The United States and Periodic Safety Reviews	Editorial changes only
14.1.5.1	The NRC's Robust and Ongoing Regulatory Process and the Current Licensing Basis	Editorial changes only
14.1.5.2	The Backfitting Process: Timely Imposition of New Requirements	Updated information on protections similar to the backfitting rule
14.1.5.3	The NRC's Extensive Experience with Broad-Based Evaluations	Editorial changes only
14.1.5.4	License Renewal Confirms Safety of Plants	No changes
14.1.5.5	Risk-Informed Regulation and the Reactor Oversight Process	Updated to describe a risk informed approach
14.1.5.6	Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations	Editorial changes only
14.1.5.7	The NRC's Regulatory Process Compared with International Safety Reviews	Updated to discuss actions to address some of the Integrated Regulatory Review Service findings
14.2	Verification by Analysis, Surveillance, Testing, and Inspection	Editorial changes only
14.3	Fukushima Lessons Learned	Update on Fukushima activities
14.4	Vienna Declaration on Nuclear Safety	New section
Article 15	RADIATION PROTECTION	No changes
15.1	Authorities and Principles	Updated and new references provided, including information on a proposed rulemaking
15.2	Regulatory Framework	Editorial changes only
15.3	Regulations	Updated status of regulatory activities and references
15.4	Radiation Protection Activities	No changes

Report Section		Change
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	Deleted outdated information on ground water contamination
15.5	Fukushima Lessons Learned	Update on Fukushima activities
Article 16	EMERGENCY PREPAREDNESS	No changes
16.1	Background	No changes
16.2	Offsite Emergency Planning and Preparedness	Editorial changes only
16.3	Emergency Classification System and Emergency Action Levels	Updated on emergency action level guidelines.
16.4	Recommendations for Protective Action in Severe Accidents	No changes
16.5	Inspection Practices - Reactor Oversight Process for Emergency Preparedness	No changes
16.6	Responding to an Emergency	Editorial changes only
16.6.1	Federal Response	Update on information on Department of Homeland Security and National Response Framework
16.6.2	Licensee, State, and Local Response	No changes
16.6.3	The NRC's Response	No changes
16.6.4	Aspects of Security that Support Response	Rewrite of actions taken after September 2011
16.7	Communications with Neighboring States and International Arrangements	Updated information on agreements and added discussion on observation of a U.S. exercise
16.8	Communications with the Public	No changes
16.9	Fukushima Lessons Learned	Updated status of Fukushima activities
Article 17	SITING	Updated to state that no changes on siting regulatory activities have taken place post-Fukushima. Addresses CNS consultancy meeting templates for Articles 17 and 18.
17.1	Background	Deleted short paragraph with duplicate information
17.2	Safety Elements of Siting	No changes
17.2.1	Background	No changes
17.2.2	Assessments of Nonseismic Aspects of Siting	Updated references
17.2.3	Assessments of Seismic and Geological Aspects of Siting	No changes
17.2.4	Assessments of Radiological Consequences from Postulated Accidents	Added updated references and editorial changes
17.3	Environmental Protection Elements of Siting	Editorial changes only
17.3.1	Governing Documents and Process	Added updated references
17.3.2	Other Considerations for Siting Reviews	Added discussion on rulemaking and other relevant regulatory developments. Minor change in section title.
17.4	Reevaluation of Site-Related Factors	Editorial changes only
17.5	Consultation with other Contracting Parties To Be Affected by the Installation	Editorial changes only

Report Section		Change
17.6	Fukushima Lessons Learned	Section deleted. No changes on siting regulatory activities have taken place post-Fukushima.
17.6	Vienna Declaration on Nuclear Safety	New section. Replaces the original 17.6.
Article 18	DESIGN AND CONSTRUCTION	Addresses CNS consultancy meeting templates for Articles 17 and 18
18.1	Defense-in-Depth Philosophy	No changes
18.1.1	Governing Documents and Process	Updated references throughout the section
18.1.2	Experience	Updates on status of Watts Bar licensing activities
18.1.2.1	Regulatory Framework for the Reactivation of Watts Bar, Unit 2	No changes
18.1.2.2	Design Certifications	No changes
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	No changes
18.3.2	Experience	References to new reactors, small modular designs, and Multinational Design Evaluation Program working groups updated
18.3.2.1	Human Factors Engineering	Cyber security rule update
18.3.2.2	Digital Instrumentation and Controls	Discussion of communication of national operating experience events and information exchanges with international counterparts
18.3.2.3	Cyber Security	Updated to discuss status of Fukushima activities
18.4	New Reactor Construction Experience Program	Discussion of communication of national operating experience events and information exchanges with international counterparts
18.5	Fukushima Lessons Learned	Updated to discuss status of Fukushima activities
18.6	Vienna Declaration on Nuclear Safety	New section
Article 19	OPERATION	Updated to state that regulatory actions post-Fukushima are discussed in Section 1
19.1	Initial Authorization to Operate	Update on new reactor licensing activities
19.2	Definition and Revision of Operational Limits and Conditions	Editorial changes only
19.3	Approved Procedures	Editorial changes only
19.4	Procedures for Responding to Anticipated Operational Occurrences and Accidents	Updated to add discussion on mitigating strategies rulemaking
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 examples of operating experience communications
19.8	Radioactive Waste	Updated to add discussion about the U.S. Court of Appeals order on Yucca Mountain
19.9	Fukushima Lessons Learned	Section deleted. Fukushima actions are discussed in Section 1.
19.9	Vienna Declaration on Nuclear Safety	New section. Replaces the original 19.9

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
APPENDICES AND ANNEXES	
APPENDIX A NRC STRATEGIC PLAN	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	Deleted. Incorporated into Fukushima writeup under Section 1.3.1.
APPENDIX C REFERENCES	Renumbered. Updated.
APPENDIX D U.S. COMMERCIAL NUCLEAR POWER REACTORS	Renumbered. Updated. Power uprates and new licenses issued.
ANNEX 2 U.S. NUCLEAR ELECTRIC INDUSTRY PERFORMANCE INDICATOR GRAPHS	Deleted

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 Sixth National Report for the Convention on Nuclear Safety," Revision 5, issued in August 2013). Lastly, this section summarizes the results of international peer reviews and missions.

1.1 The U.S. Policy toward Nuclear Activities

The Energy Reorganization Act of 1974 created the U.S. 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 NRC's 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's 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, predecisional information, and 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 also has 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 NUREG-1614, Volume 6, "Strategic Plan: Fiscal Years 2014-2018," dated August 2014. 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 current and expected future challenges include:

- continued implementation of enhancements to nuclear safety based on insights arising from operating experience reviews and lessons learned from the 2011 nuclear accident at the Fukushima Dai-ichi nuclear facility in Japan

- continual learning and adaptation of the regulatory framework, as necessary, to address knowledge of and response to the specific hazards, uncertainties, and risks associated with each nuclear site

- continued readiness to review applications involving new technologies such as small modular reactors, medical isotope production facilities, and rapidly evolving digital instrumentation and control systems

- changes in the demographics, experience, and knowledge of the workforce

- continued awareness of and support to the development of nuclear safety and security regulations around the world

- changing economic conditions in the energy market affecting current and planned applications to construct and operate new nuclear facilities or licensee decisions to decommission existing ones

- globalization of nuclear technology and the nuclear supply chain, driving the need for increased international engagement on the safe and secure use of radioactive material and the need for new oversight approaches, including ensuring that foreign components used in U.S. nuclear facilities are in compliance with NRC requirements

- continuous monitoring of the threat environment to ensure the security of nuclear facilities and radioactive materials

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

- market pressures on operating plants and license applications
- a significant operating incident (domestic or international)
- globalization of the nuclear technology and the nuclear supply chain
- a significant terrorist incident
- legislative and executive branch initiatives
- international nuclear standards developments
- international treaties and conventions
- lost, misplaced, intercepted, or delayed information

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. These challenges represent what the Inspector General considers to be inherent and continuing program challenges relative to maintaining effective and efficient oversight and internal controls. As a result, it is likely they will continue to be challenges from year to year. Challenges do not necessarily equate to problems. The 2015 assessment report described the main challenges in the following areas of NRC's work, discussed in more detail in Appendix B to this report.

- regulation of nuclear reactor safety programs

- regulation of nuclear materials and radioactive waste programs

- management of security over internal infrastructure (personnel, physical, and cyber security) and nuclear security

- management of information technology and information management

- management of financial programs

- management of administrative functions

1.2 National Nuclear Programs

The NRC has several 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 in the reactor arena comprise a well-established regulatory process, which includes: (1) reactor oversight, (2) license renewal, (3) power uprates, and (4) new reactor licensing.

1.2.1 Reactor Oversight Process

The regulatory framework for NRC's Reactor Oversight Process consist of three strategic performance areas: reactor safety, radiation safety, and safeguards. Within each strategic performance area are cornerstones that reflect the essential safety aspects of facility operation. These seven cornerstones include: initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, and physical security. Satisfactory licensee performance in the cornerstones provide reasonable assurance of safe facility operation and that the NRC's safety mission is being accomplished. Each cornerstone contains performance indicators to ensure that their objectives are being met.

The results of the annual self-assessments and other independent or focused evaluations have stated that the risk-informed performance-based Reactor Oversight Process has remained transparent and showed that the Reactor Oversight Process has effectively and openly supported the agency's mission and strategic goals of safety and security.

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 specifically 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, more than 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 sixth U.S. National Report, 18 units entered their 41st year of operation (the period of extended operation) between 2011 and 2013.[2] By the end of 2016, 20 additional units will have entered the period of extended operation as listed below. In addition to these plants, Indian Point Nuclear Generating Station, Units 2 and 3, are operating beyond 40 years under the timely renewal provision.[3]

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2 In October 2015, Pilgrim Unit 1, which entered the period of extended operation in 2012, announced that the plant will cease operations by mid-2019. In April 2016, Entergy announced that Pilgrim will be permanently shutdown on May 2019.

3 Visit http://www.nrc.gov/info-finder/reactors/ip/ip-timely-renewal.html for more information on Indian Point's timely renewal.

Year 2014	Year 2015	Year 2016
• Calvert Cliffs Nuclear Power, Unit 1 • Oconee Nuclear Station, Unit 3 • Arkansas Nuclear One, Unit 1 • Edwin I. Hatch Nuclear Plant, Unit 1 • Peach Bottom Nuclear Plant, Unit 3 • Donald C. Cook Atomic Power Station, Unit 1 • Browns Ferry Nuclear Plant, Unit 2 • Brunswick Steam Electric Plant, Unit 2 • James A. FitzPatrick Nuclear Power Plant[4] • Three Mile Island Nuclear Station, Unit 1 • Cooper Nuclear Station • Duane Arnold Energy Center • Prairie Island Nuclear Generating Plant, Unit 2	• Millstone Power Station, Unit 2	• Beaver Valley Power Station, Unit 1 • St. Lucie Nuclear Power Plant, Unit 1 • Browns Ferry Nuclear Plant, Unit 3 • Calvert Cliffs Nuclear Power Plant, Unit 2 • Salem Nuclear Generating Station, Unit 1 • Brunswick Steam Electric Plant, Unit 1

Section 1.3.2 of this report provides a discussion on subsequent license renewal (i.e., renewal beyond 60 years). Section 6.2 of this report provides additional discussion on the Pilgrim Nuclear Power Station, Unit 1, and James A. FitzPatrick Nuclear Power Plant license status. Article 14 of this report discusses the license renewal process in detail, including a discussion of the update to the Generic Environmental Impact Statement for license renewal.

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, the 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

4 FitzPatrick entered the period of extended operation in October 2014. However, in March 2016, Entergy certified that FitzPatrick would cease operations in January 2017.

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's new reactor program focuses on licensing reviews for small and large light-water reactors and advanced nonlight-water reactors; oversight and construction inspection activities; preapplication and readiness reviews for current and future reactor licensing; and infrastructure development to support oversight and reactor licensing. The NRC is in the process of completing ongoing licensing reviews; supporting construction activities associated with four new reactor units in the United States licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"; and is making enhancements to increase the efficiency and predictability of small light-water and advanced reactor reviews. 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 four combined license applications for a total of seven new nuclear plants, two design certification applications, and one design certification renewal application. All combined license applicants are using the licensing process specified in 10 CFR Part 52. 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.

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, Hungary, India, Japan, the Republic of Korea, the Russian Federation, South Africa, Sweden, the United Arab Emirates, the United Kingdom, and the United States. 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 Advanced Passive (AP) 1000, Korea Electric Power Corporation and Korea Hydro and Nuclear Power Co., Ltd.'s Advanced Power Reactor 1400 (APR1400), General Electric Nuclear Energy's Advanced Boiling-Water Reactor, and AREVA Nuclear Power's U.S. Evolutionary Power Reactor (US 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.

Articles 17 and 18 of this report discuss new reactor licensing in more detail. Sections 1.3.3, 6.2, and 18.1.2 of this report discuss licensing of Watts Bar, Unit 2, 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 sixth U.S. National Report and addresses those safety and regulatory issues and regulatory accomplishments that have necessitated significant attention since the last National Report was issued.

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

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

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

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:

- 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>Shield Building Laminar Cracking at Davis-Besse</u>

The shield building at the Davis-Besse nuclear power plant is a reinforced-concrete structure that surrounds the freestanding steel containment vessel, and has nominal wall thickness of 30 inches with vertical and horizontal rebar grids on both the inside and outside face. The functions of the shield building are to provide: (1) biological shielding, (2) environmental protection of the containment vessel, and (3) control release of annulus atmosphere during accidents.

During the October 2011 mid-cycle outage, while cutting a construction opening to replace the reactor vessel closure head, laminar cracking was identified in the shield building cylindrical wall. The licensee performed impulse response mapping, core bores, and in-depth calculations to determine the operability of the shield building. 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 the reason for the concrete laminar cracking was a combination of the design specification for construction of the shield building, which did not require application of an exterior sealant from moisture, and environmental conditions associated with the blizzard of 1978.

The NRC issued "Confirmatory Action Letter – Davis-Besse Nuclear Power Station," on December 2, 2011, detailing planned actions that the licensee had to take to provide continued long-term confidence of the shield building's ability to maintain its safety functions. The licensee performed structural evaluations to capture bounding conditions and took corrective actions including (1) applying a sealant to the surface of the shield building to address the cause of the cracking and prevent new cracks and (2) development of a shield building monitoring program.

In the summer of 2013, while performing additional testing according to the shield building monitoring program, the licensee identified new crack indications. The licensee determined that some of these cracks were previously missed because of limitations of the boroscope used for earlier examinations of core bores and others were considered evidence of possible growth of the existing cracks. The licensee performed more testing and analysis that confirmed crack growth and issued an apparent cause evaluation that attributed the crack growth to ice wedging. The licensee modified the shield building monitoring program in response to these new findings.

The NRC has completed an inspection of the licensee's apparent cause evaluation for the crack growth. The NRC staff determined that the shield building laminar cracking condition remained bounded by the licensee's structural evaluation. The NRC has concluded that the licensee provided reasonable assurance that, with the current condition, the shield building will perform its safety function, including withstanding earthquakes and tornadoes. The NRC staff continues to follow the licensee's corrective actions related to this issue under the baseline inspection program.

<u>Alkali-Silica Reaction Concrete Degradation at Seabrook Station</u>

Alkali-silica reaction is a slow chemical process that can cause degradation 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 microcracks 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 concrete mechanical properties assumed in the American Concrete Institute design and construction codes. Alkali-silica reaction expansion could also lead to structural displacement or deformation and discrete macrocracking not considered in the concrete design, and could 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 U.S.

commercial nuclear power plant where this type of degradation has been identified. 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 used 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 engineers have continued detailed testing, walkdowns, crack monitoring, and evaluations to address and manage the issue comprehensively in the short- and long-term. On May 16, 2012, the NRC staff issued a letter to the licensee to confirm commitments to 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, in part, to address long-term effects on structural performance and management of the issue, and to provide a technical basis for resolution of the operability determination and for identifying 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, verified the adequacy of planned actions related to the alkali-silica reaction structures monitoring program, and the large-scale testing to reconcile this issue with the design and licensing basis. The large-scale testing has been completed and the licensee is currently compiling and reviewing the results of the testing program in preparation of a license amendment request submittal. An initial determination of the test program is that out-of-plane (through-thickness) expansion is also an important parameter for monitoring the progression of alkali-silica reaction. The licensee is in the process of installing extensometers at select plant locations to measure future out-of-plane expansion.

The NRC also has 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 Information Notice (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 slow nature of the degradation, and ongoing monitoring. This review has included an evaluation of the licensee's prompt operability determinations for various structures affected by alkali-silica reaction. These operability determinations address the alkali-silica reaction impacts on material properties due to microcracking, as well as the impacts due to macrocracking and building deformation due to alkali-silica reaction expansion. The NRC's oversight includes ongoing assessment of the continued acceptability of the operability determinations. The most recent results of the NRC staff's review are documented in "Seabrook Station, Unit No. 1 – Integrated Inspection Report 05000443/2015004 and Independent Spent Fuel Storage Installation Report No. 07200063/2015001," dated February 12, 2016. The NRC continues its oversight of the alkali-silica reaction issue and has formed a multioffice, multidiscipline working group to guide the agency's ongoing approach to respond to this safety issue.

Construction Inspection Program Lessons Learned

To provide regulatory oversight of the construction of four AP1000 units, NRC implemented a construction inspection program, including development of governing documents and procedures. The NRC's Region II Office in Atlanta, Georgia, has the primary responsibility for implementing the construction inspection program. Region II has as many as five resident construction inspectors at the construction sites during the preoperational phase of construction to oversee the day-to-day activities of the licensee and its contractors, and supplements 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.

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 evaluated inspection results to identify lessons learned that could 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 several years 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 aware of and 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.

- Corrective Actions —Implementing an effective corrective action program at construction sites presents unique challenges to licensees that have a large number of activities occurring across many organizations (licensee, contractors, vendors, etc.).

- 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.

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 their associated suppliers. Although there is no evidence of significant counterfeit activity impacting U.S. nuclear facilities, the NRC emphasizes the importance of robust quality assurance programs in protecting against counterfeit, fraudulent, and suspect items.

Over the past 3 decades, the NRC has published multiple documents to inform stakeholders of counterfeit or misrepresented products and services. On March 21, 1989, the NRC issued Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," to inform licensees of effective program elements for detecting counterfeit or fraudulently marketed products and for assuring the quality of vendor-supplied products.

More recently, the NRC has published several other documents related to counterfeit, fraudulent, and suspect items:

- IN 2008-04, "Counterfeit Parts Supplied to Nuclear Power Plants," dated April 7, 2008, informs addressees of the potential for counterfeit parts to enter their supply chains.

- IN 2012-22, "Counterfeit, Fraudulent, Suspect Items (CFSI) Training Offerings," dated January 25, 2013, provides a list of training resources that can be used for educating personnel involved in NRC-regulated activities on current trends in CFSI and techniques to prevent the use of CFSI.

- IN 2013-15, "Willful Misconduct/Record Falsification and Nuclear Safety Culture," dated August 23, 2013, describes a vendor's criminal actions to destroy serial numbers in an attempt to conceal a component's origin before it was installed in a U.S. nuclear plant.

- Regulatory Issue Summary (RIS) 2015-08, "Oversight of Counterfeit, Fraudulent, and Suspect Items in the Nuclear Industry," dated June 24, 2015, heightens awareness of the existing NRC regulations and how they apply to counterfeit, fraudulent, and suspect items within the scope of the NRC's regulatory jurisdiction.

- Regulatory Guide (RG) 5.71, "Cyber Security Programs for Nuclear Facilities," dated January 2010, addressed the procurement of digital assets including supply chain security.

Licensees and industry organizations have also focused on the challenges presented by counterfeit, fraudulent, and suspect items. In a joint effort with the Nuclear Energy Institute, the Electric Power Research Institute (EPRI) developed EPRI-1019163, Revision 1, "Plant Support Engineering: Counterfeit and Fraudulent Items," in July 2014. The guidance document is intended for use by licensees to aid in preventing the introduction of counterfeit, fraudulent, and suspect items into nuclear facilities.

The NRC has also been involved in other U.S. Government activities related to addressing counterfeit, fraudulent, and suspect items. The NRC participates in the U.S. Department of Homeland Security's National Intellectual Property Rights Coordination Center, which leverages the combined resources of partner agencies to better combat intellectual property theft and to dismantle the criminal organizations that seek to profit from the manufacturing, importation and sale of counterfeit items.

To consolidate information and references for its activities related to counterfeit, fraudulent, and suspect items, the NRC developed an extensive public Web site, which can be accessed at http://www.nrc.gov/about-nrc/cfsi.html. The information shared on the Web site includes historical agency documents, presentations from various public meetings, and ongoing NRC activities to address counterfeit, fraudulent, and suspect items.

Cumulative Effects of Regulation

In 2009, the NRC began addressing ways to mitigate the cumulative effects of regulation in response to Commission direction. Since then, the staff prepared four Commission papers: SECY-11-0032, "Consideration of the Cumulative Effects of Regulation in the Rulemaking Process," dated March 2, 2011, SECY-12-0137, "Implementation of the Cumulative Effects of Regulation Process Changes," dated October 5, 2012, COMSECY-14-0014, "Cumulative Effects of Regulation and Risk Prioritization Initiative: Update on Recent Activities and Recommendations for Path Forward," dated April 9, 2014, and SECY-15-0050, "Cumulative Effects of Regulation Process Enhancements and Risk Prioritization Initiative," dated April 1, 2015.

The Commission approved the rulemaking enhancements proposed by the staff in SECY-11-0032, which included providing increased stakeholder interactions, publishing supporting guidance concurrent with rules, requesting specific comment on cumulative effects of regulation process improvements in proposed rules, and developing informed implementation timeframes.

In response to SECY-12-0137, the Commission directed the following:

- 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 the NRC’s regulatory analysis.

In COMSECY-14-0014, the staff provided a summary of the results of case studies to review the accuracy of cost and schedule estimates used in the agency’s regulatory analyses. In addition, the staff proposed to merge the cumulative effects of regulation and risk prioritization initiative deliverables, which was approved by the Commission. In SECY-15-0050, the staff provided options to the Commission for incorporating risk insights into the decisionmaking process to prioritize regulatory activities for operating reactors. In response to SECY-15-0050, the Commission continued to support ongoing cumulative effects of regulation process enhancements, including consideration of risk insights in regulatory decisionmaking through existing agency processes.

In addition, the NRC is applying the process enhancements principles, 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 the development of these actions, and is providing implementation guidance when necessary.

Evaluation of Economic Consequences

The NRC’s regulatory framework accounts for the offsite economic consequences associated with unintended releases of radionuclides with subsequent land contamination. Specifically, offsite property damage is considered during the evaluation of cost-justified substantial safety enhancements (i.e., backfit analysis), as well as in regulatory and environmental analyses. The NRC uses similar guidance documents to conduct the cost-benefit determinations of these analyses. In performing these economic analyses, the NRC has traditionally considered two categories of property, onsite and offsite. Generally, onsite property is owned or controlled by the license- or certificate-holder and located within the boundaries of the licensed facility, whereas offsite property is located outside of the site boundaries, and is not owned or controlled by the license- or certificate-holder. However, in these economic analyses, the distinction between offsite and onsite property goes beyond the location or ownership of the property. Onsite property costs include replacement power, decontamination costs, and costs associated with refurbishment or decommissioning. Offsite property costs include both the direct costs associated with property damage (e.g., diminution of property values) and indirect costs (e.g., tourism, manufacturing, and agriculture disruption).

In response to SECY-12-0110, “Consideration of Economic Consequences Within the U.S. Nuclear Regulatory Commission’s Regulatory Framework,” the Commission directed the NRC staff to enhance the currency and consistency of the existing framework through updates to guidance documents integral to performing cost-benefit analyses. The Commission also found that economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety. Therefore, the Commission does not plan to discuss economic consequences in future CNS reports.

Fukushima Lessons Learned

After the accident at Fukushima, the NRC took prompt action to ensure that there were no immediate safety concerns at U.S. facilities and to verify 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 provide information about the Fukushima accident to U.S. licensees. The NRC also issued two inspection procedures (IPs), called temporary instructions (TIs), to NRC inspection staff 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 the 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, titled “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 to public health and safety. The NTTF also concluded that enhancements to safety and emergency preparedness were warranted, and made 12 overarching recommendations for Commission consideration. In addition, the NTTF concluded that nuclear power plant facilities and designs that were under NRC review, or licensed, since 2011, also needed to demonstrate their ability to cope with hazards that are beyond the design basis. The staff proposed a prioritization of the NTTF’s recommendations. The Commission approved the staff’s proposal in SRM-SECY-11-0137, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned,” dated December 15, 2011, and stated its support for staff action on the near-term recommendations.

The NRC formed the Japan Lessons-Learned Project Directorate (currently called the Japan Lessons Learned Division) to perform a longer-term review of the March 11, 2011, Japanese earthquake and tsunami and lead the implementation of the associated safety enhancements. This organization reports to a Steering Committee of senior NRC officials, which is chaired by the Deputy Executive Director for Reactor and Preparedness Programs and it is comprised of Office Directors from many of the NRC program offices and regional offices. As a first step in its assessment of lessons-learned from the accident, the staff considered whether any of the NTTF recommendations identified an imminent hazard to public health and safety. The staff ultimately agreed with the NTTF’s conclusion that the accident did not reveal any imminent risk to public health and safety. The Japan Lessons-Learned Project Directorate then prioritized the twelve NTTF recommendations by tiers and expanded upon the task force recommendations to include proposals from the international community, the U.S. Congress, the NRC’s Advisory Committee on Reactor Safeguards, and other stakeholders.

Tier 1 Recommendations

The first tier consists of actions that the NRC determined should be started without unnecessary delay and for which sufficient resource flexibility, including availability of critical skill sets, exists. The Tier 1 recommendations consist of the following:

- 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*)
- spent fuel pool (SFP) instrumentation (*Recommendation 7.1*)
- strengthening and integration of emergency operating procedures, 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 were recommendations that the staff determined did not require long-term study and could be initiated when sufficient technical information and applicable resources become available. The Tier 2 recommendations include the following:

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

As a result of further assessment after these items were prioritized as Tier 2 recommendations, the NRC determined that SFP makeup capability and the emergency preparedness actions would be more efficiently and effectively 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.

In SECY-16-0074, “Assessment of Fukushima Tier 2 Recommendation Related to Evaluation of Natural Hazards Other Than Seismic And Flooding,” dated June 2, 2016, the staff concluded that other than seismic and flooding, only those natural hazards associated with high winds and snow loads warranted further assessments and stakeholder interactions to address the recommendation. The staff intends to complete its assessment by the end of 2016.

Tier 3 Recommendations

The third tier consists of actions that required further staff study to support regulatory action, relied on the result of an associated short-term action to inform the long-term action, depended on the availability of critical skill sets, or related to potential revisions to the regulatory framework that balances defense-in-depth and risk considerations (Recommendation 1). The following recommendations are included in Tier 3:

- periodic 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 multiunit events (dependent on availability of critical skill sets) (*Recommendation 9.1/9.2*)
- Emergency Response Data System capability (related to long-term evaluation under Recommendation 10) (*Recommendation 9.3*)
- additional emergency preparedness topics for prolonged SBO and multiunit 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 SAMGs (dependent on Recommendation 8) (*Recommendation 12.2*)
- basis of emergency planning zone size (*additional issue*)
- prestaging of potassium iodide beyond 10 miles (*additional issue*)
- expedited transfer of spent fuel to dry cask storage (*additional issue*)

The evaluation of the need to expedite transfer of spent fuel to dry cask storage was addressed by the staff in COMSECY-13-0030, “Staff Evaluation and Recommendation for Japan

Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 12, 2013. The Commission agreed with the staff's assessment that expedited transfer was not warranted. In addition, a number of Tier 3 recommendations are being addressed by the Tier 1 "Mitigation of Beyond-Design Basis Events" rulemaking, which can be found in www.regulations.gov (Docket ID: NRC-2014-0240). In SECY-15-0137, "Proposed Plans for Resolving Open Fukushima Tier 2 and 3 Recommendations," dated October 29, 2015, the staff proposed resolution plans, including necessary resource requirements, for those Tier 2 and 3 recommendations that had not been previously closed or addressed by other, higher-priority recommendations. Staff Requirements Memorandum (SRM)-SECY-15-0137, dated February 8, 2016, approved the staff's resolution plans for these Tier 2 and 3 recommendations. The Commission approved closing a subset of these recommendations and directed the staff to continue issuing status updates on those recommendations that remain open.

Subsequently, in SECY-16-0041, "Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, And Enhanced Instrumentation," dated March 31, 2016, the staff informed the Commission of the final assessment and closure of Fukushima-related Tier 3 recommendations regarding evaluations of reliable vents for containment types other than BWR Mark I and Mark II containments, hydrogen control and mitigation, and reactor and containment instrumentation enhancements.

Post-Fukushima Safety Enhancements

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

On March 12, 2012, the NRC issued the first regulatory requirements, in the form of orders, for the operating reactors based on lessons learned from the accident. 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 beyond-design-basis external events, (2) ensuring severe-accident-capable reliable hardened containment vents for boiling-water reactors (BWRs) with Mark I and II containment designs, and (3) enhancing SFP instrumentation. All licensees will be in compliance with the SFP instrumentation order, and the majority of the licensees will be in compliance with the mitigating strategies order, by the end of 2016.

The NRC also issued an RFI on March 12, 2012, requiring each reactor licensee to reevaluate the seismic and flooding hazards at its site using present-day guidance, 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. The walkdowns have been completed. The NRC has received the reevaluated hazard reports for seismic and flooding for most licensees and has begun to issue assessments of those reports. As appropriate, licensees are performing additional evaluations to determine how the reevaluated hazards would affect the plant. They are also assessing whether the mitigating strategies developed in response to the order described above would remain available under the conditions of the reevaluated hazards. If not, then under the proposed related rulemaking, they would be required to enhance those strategies, develop alternate mitigating strategies, or, possibly, make design basis changes under the NRC's backfit provisions.

Information about the NRC's 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 in July 2012. Specifically, per paragraph 23 of the Final Summary Report for the 2nd CNS Extraordinary Meeting, the Contracting Parties agreed that the National Reports should cover:

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

The NRC is undertaking near-term regulatory activities to reevaluate and enhance, as necessary, the protection of structures, systems, and components (SSCs) against 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 senior management and Commission 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 that exceed a facility's design basis. Activities associated to these topics include the following:

- evaluation of external hazards other than seismic and flooding
- updating of natural hazard information and addressing any new and significant information
- development of mitigating strategies for beyond-design-basis external events

Additional details are discussed in Section 1.3.3 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 offsite contamination.

In response to the Fukushima accident, the NRC issued an order requiring all licensees to have mitigating strategies that would preserve core cooling, SFP cooling, and containment. The NRC issued the Mitigation of Beyond-Design-Basis Events proposed rulemaking that would make this requirement generally applicable and incorporate lessons learned from implementation of the order. Furthermore, 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. Information from these evaluations will be used to determine whether additional regulatory actions are necessary (e.g., updating the design-basis and SSCs important to safety) to protect against the updated hazards. Additional details are discussed in Section 1.3.3 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. The development of probabilistic safety assessments to identify additional accident management measures are discussed in Section 10.3.5 of this report.

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

As noted in Section 8.3 of this report, the U.S. Congress created the NRC as an independent agency in 1974. 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 the International Atomic Energy Agency's (IAEA) peer review missions, the United States hosted an IRRS mission in 2010 and the followup mission in 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 multiunit 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.

The Fukushima accident highlighted the need to ensure that sufficient staff and resources are available to respond to a multiunit event and a prolonged SBO. As such, the NRC undertook actions to enhance emergency preparedness with respect to communications and staffing. 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 and regulatory guidance. 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 emergency response events. Additional information can be found in Sections 8.1.7 and 8.3 of this report.

The NRC staff has also continued to support and participate in NEA’s post-Fukushima activities, notably through its Committee on Nuclear Reactor Activities and Committee on the Safety of Nuclear Installations. The NRC has been involved in a number of working groups, including those associated with defense-in-depth and defining an effective nuclear regulator, along with research activities. The NRC staff considers the NEA guidance and research results for harmonization with its actions. The NRC will continue to support the NEA’s long-term post-Fukushima research activities and will consider the results for harmonization with its planned actions.

The NRC has long been involved with the international community in collaborative efforts related to safety analysis and assessment. Since the Fukushima accident, international cooperation has been strengthened further through safety related projects coordinated by NEA, International Atomic Energy Commission, and other regional organizations. The Benchmark Study of the Accident at the Fukushima Dai-ichi Nuclear Power Station (BSAF) project under the auspices of NEA is an example of such cooperative effort. The U.S. Department of Energy (DOE), the NRC, EPRI, and others participate in this benchmark phased study. The main objectives of the project are to benchmark severe accident analysis tools (codes) against the Fukushima Dai-ichi accident, and to help Japan prepare for decommissioning of Fukushima Dai-ichi, Units 1, 2, and 3, by using the codes to predict core debris locations to better inform approaches for access and retrieval of radioactive material. Phase 1 of the project focused on thermal-hydraulics of the reactor systems and the containment, and an estimation of the distribution of degraded core materials and their composition for the first 6 days of the accident. Phase 1 was completed and the final report was issued in March 2015. Phase 2 started in April 2015 with the goal of analyzing fission product transport and distribution on-site and off-site for the first 21 days of the accident. It is estimated that Phase 2 will continue through mid-2018.

Another Fukushima-related safety project under the auspices of NEA is SAfety REsearch Opportunities post-Fukushima (SAREF). The project has two objectives: (1) to address safety research gaps and advance safety knowledge base, and (2) to support Japan in achieving safe and prompt decommissioning. A report summarizing the safety research gaps is in preparation. NEA is also sponsoring a study of fast-running software tools used to model radionuclide releases during nuclear accidents in the frame of the Fukushima Dai-ichi accident. The objective of this activity is to benchmark software tools used to estimate consequences of accidents at nuclear facilities. A report summarizing the results of these studies will be presented for approval during the NEA’s Committee on Nuclear Regulatory Activities (CNRA) meeting in June 2016.

The NRC's post-Fukushima activities are also consistent with the IAEA's report titled "Action Plan on Nuclear Safety," which was adopted by the General Conference in September 2011. The report identified specific initiatives to address lessons learned from the nuclear accident and to enhance multilateral communication. 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 operating nuclear power plants as discussed in this section and Section 1.3.3 of this report.

- 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 IRRS and Operational Safety Review Team (OSART) missions around the world, often at a senior leadership level. Additional information can be found in Section 1.4 of this report.

- Emergency Preparedness and Response: The United States has undertaken significant activities to assess and strengthen, where appropriate, its emergency preparedness and response programs following the Fukushima accident. The United States has also worked closely with Canada and Mexico to enhance North American cooperation in this area. Additional information can be found in Sections 16.9 and 16.7 of this report.

- National Regulatory Bodies: The NRC has devoted significant resources to address the findings and recommendations from the U.S. IRRS mission. A thorough assessment of the U.S. nuclear safety regulatory infrastructure and current regulations was a key component in the NRC's assessment methodology. Additional information can be found in Section 8.1.5.2 of this report.

- Operating Organizations: The licensee has primary responsibility for safety in the United States. The U.S. Government continues to work closely with INPO to ensure clear communication with each of our licensees and has directed each licensee to implement recommendations from nuclear safety assessments. The United States also continues to host OSART missions on a regular basis. Additional information can be found in this section and in Sections 1.3.3 and 8.1.5.3, Article 9, and Part 3 of this report.

- IAEA Safety Standards: Through its representation on the IAEA Commission on Safety Standards and all of the IAEA 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 and regulatory guidance. Additional information can be found in Section 8.1.5.1 of this report.

- International Legal Framework: The United States conducts international activities related to statutory mandates, international treaties and conventions, international organizations, bilateral relations, and research. For example, in this report, the United States has endeavored to address, in detail, all the areas specified in the revised CNS guidance documents, Rapporteur's findings, and the Vienna Declaration on Nuclear Safety. The United States has encouraged other contracting parties to do likewise through its bilateral and multilateral activities. Additional information can be found in Sections 1.4.1 and 8.1.5 of this report.

- Member States Planning to Embark on a Nuclear Power Program: 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. Additional information can be found in Section 8.1.5 of this report.

- 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. Additional information can be found in Sections 1.3.2, 8.1.6.2 and 11.2 of this report.

- Protection of People and the Environment from Ionizing Radiation: The United States, through its regulatory framework, regulations, and radiation protection programs, continues to ensure that radiation exposure to the workers caused by a nuclear installation is kept as low as reasonably achievable, and that the potential for exposures of individuals to radiation doses that exceed the prescribed national dose limits are minimized. Additional information can be found in Section 15 of this report.

- Communication and Information Dissemination: The United States places a high priority on effective and transparent communication with the public in the event of an emergency. Additional information can be found in Section 16.8 of this report. Information about the NRC's openness and transparency policies and practices can be found in Section 8.1.7 of this report.

- 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. Additional information can be found in Section 6.3.10 of this report.

The NRC's post-Fukushima activities are consistent with the IAEA Fukushima Report. Furthermore, the NRC activities address, as appropriate, the observations and challenges identified by the Special Rapporteur on Fukushima, which are documented in the 6th CNS President's report. Specifically, the Special Rapporteur identified the following challenges:

- Minimize Gaps Between Contracting Parties' Safety Improvements: Through its representation on the IAEA 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 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. The NRC also provides senior expert assistance to the IAEA to support studies designed to advance the safety standards program and minimize gaps in safety improvements. Additional information can be found in Section 8.1.5.1 of this report.

- Achieve Harmonized Emergency Plans and Response Measures: After the Fukushima event, the NRC undertook actions to enhance emergency preparedness for licensees with respect to communications and staffing. For example, the NRC requested that licensees evaluate their current communications systems and the equipment that would be used during an emergency event. The NRC also improved its communication strategy and conducted an extensive analysis of the emergency planning zone. Additional details can be found in Section 16.9 of this report.

- Make Better Use of Operating and Regulatory Experience, and International Peer Review Services: The United States supports IAEA peer review services, such as IRRS and OSART missions. The NRC also has a robust operating experience program, which is considered essential in the implementation of the agency's mission. Additional information can be found in Sections 1.4 and 6.3.5 of this report.

- Improve Regulators' Independence, Safety Culture, Transparency and Openness: The United States places a high priority on effective and transparent communication with the public in the event of an emergency. After the Fukushima event, the NRC updated its crisis communication plan to support public and media outreach efforts in future emergency response eventbas. The actions that the NRC has taken are broadly supportive of the priority given to safety and a strong safety culture. For example, the NRC has ensured that licensees have adequate staffing to respond to emergencies and clearly defined roles and responsibilities for those responders. The NRC also developed a Safety Culture Policy Statement, which applies to all of its regulated entities. The NRC has updated all appropriate guidance and inspection documents to address the Safety Culture concepts in the Policy Statement. Additional information can be found in this section and Sections 1.3.3, 8.1.7, 8.3, and 10.4 of this report.

- Engage All Countries to Commit and Participate in International Cooperation: The United States has undertaken significant activities to assess and strengthen, where appropriate, international cooperation. The United States has worked closely with Canada and Mexico to enhance North American cooperation in this area. The United States also continues to lead by example in the implementation of conventions and treaties, such as the CNS. The United States has encouraged other contracting parties to do likewise through its bilateral and multilateral activities. Additional information can be found in Sections 1.4.1, 8.1.5, 16.9, and 16.7 of this report.

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 and regulatory practices. In other cases, the failure of nondestructive evaluation to find significant degradation has led to updated approaches to improve the performance of the evaluation. Several examples of the implications of these nondestructive evaluation issues are provided below.

Failed Inspections of Diablo Canyon, Unit 2, Weld Overlays

In 2008, Diablo Canyon, Unit 2, installed Alloy 52 full structural weld overlays on six pressurizer welds. The inspection vendor performed an acceptance examination using conventional manual ultrasonic search units to detect possible welding defects in the overlays. The acceptance review found no unacceptable indications.

In 2013, the licensee performed an inservice inspection using a manual phased array ultrasonic search unit. This examination found several embedded laminar flaws missed by the 2008 acceptance examinations. One of the flaws was 16 inches long and a second was essentially 360 degrees around the overlay. It was determined that the IP and search units used in the 2008 acceptance examinations were able to detect the laminar flaws, but the procedure had been implemented poorly. The original ultrasonic search units were difficult to hold perpendicular to the pipe at high scanning speeds and procedures failed to address this limitation. The phased-array search units were contoured to fit the pipe surface and were easier to use. Additionally, the phased-array search units provide a more persistent image of the flaws, allowing the inspector more time to detect and identify the flaws.

After a structural analysis was performed by the licensee and the analysis was reviewed by the NRC staff, it was determined that the laminar flaws did not challenge the structural integrity of the weld overlays. No repairs were required.

The licensee has implemented several changes to the weld overlay inspections. The licensee is disallowing the use of the 2008 IP in future inspections. The licensee will only employ phased array probes for subsequent examinations of pressurizer weld overlays. The IP used in the 2008 examination was revised by EPRI to provide additional guidance for the maximum scan speed.

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, 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.

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, reviewed the licensee fracture evaluation and causal analysis and determined that Doel, Unit 3, and Tihange, Unit 2, would be permitted to restart. The Federal Agency for Nuclear Control has placed a number of conditions on the restart of both reactors. Among these conditions, the licensee was required to perform irradiation studies on some surrogate materials. In 2014, the results of the irradiation studies became available and indicated the surrogate materials experienced more embrittlement than expected. As a result, Doel, Unit 3, and Tihange, Unit 2, entered into early refueling outages and remained shut down due to pending investigation of anomalous embrittlement. The licensee performed additional irradiation studies and determined the anomalous embrittlement was a function of the surrogate material, and was not representative of the actual reactor pressure vessel forging material. As a result, the Federal Agency for Nuclear Control permitted Doel 3 and Tihange 2 to resume operations.

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 retrieved the original fabrication records of ultrasonic examinations of vessel ring forgings and documented the ability of the construction-era ultrasonic examination techniques to detect indications similar in nature to hydrogen flakes, and documented the requirement for recording such indications. The industry also performed bounding structural integrity assessments to ensure that reactor pressure vessel integrity would be maintained under accident conditions, even in the presence of hydrogen flaking.

On September 22, 2013, the NRC issued IN 2013-19, "Quasi-Laminar Indications in Reactor Pressure Vessel Forgings," to inform industry of the quasi laminar indications observed in the Belgian reactor pressure vessel forgings. Additionally, the NRC hosted a public meeting with industry and stakeholders on March 5, 2013, to discuss these indications. The industry presented plans to investigate the type of ultrasonic examination techniques used during construction and to perform a probabilistic fracture mechanics evaluation of the structural integrity effect on U.S. reactors of potentially undiscovered quasi laminar indications.

Subsequently, in October 2013, EPRI published its findings in EPRI-3002000647, "Materials Reliability Program (MRP): Evaluation of the Reactor Vessel Beltline Shell Forgings of Operating U.S. PWRs [Pressurized Water Reactors] for Quasi Laminar Indications (MRP 367)." The objectives of the report were twofold: (1) to evaluate whether reactor pressure vessel forgings in U.S. plants were likely to have indications similar to those found in Doel 3 and Tihange 2, and (2) to evaluate the structural significance of indications if they did exist in a reactor pressure vessel. The report concluded that the ultrasonic techniques used during construction of U.S. vessels were capable of detecting quasi laminar indications, and the reporting requirements would have caused the indications to be recorded if they were present. The report included a probabilistic fracture mechanics analysis of a set of conditions based on data from Doel 3 and Tihange 2. The industry concluded that even if quasi laminar indications were present in a U.S. reactor vessel forging, the incremental increase in the vessel failure probability under pressurized thermal shock loading is negligible.

The NRC staff's evaluation consisted of reviewing the analyses performed by the Belgian licensee as well as the evaluation performed by the industry. Specifically, the NRC staff reviewed evaluations of nondestructive examination records performed by the U.S. industry to examine the likelihood of the presence of the quasi laminar indications in U.S. reactor pressure vessels. Structural evaluations were also performed to determine the risk significance even if the quasi laminar indications were present. This was followed by a risk-informed evaluation documented in "Technical Assessment of Potential Quasi-Laminar Indications in Reactor

Pressure Vessel Forgings," dated September 8, 2015. 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 American Society of Mechanical Engineers (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.

Identification of Cracking in a Bottom Mounted Instrument Nozzle at a French Plant

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 Boiler and Pressure Vessel Code (BPV Code). The BPV Code currently only requires visual examination of the outer surface of the reactor vessel where the bottom mounted nozzles exit the lower head. Because of the nondestructive evaluation findings at the French plant, the NRC staff approached ASME to initiate BPV 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 ASME Code committee considered developing a code case, but found only a limited number of licensees would consider using the code case. Therefore, the committee elected not to develop a code case. Licensees who want to perform volumetric inspections in lieu of visual inspections can do so using the NRC's relief request process.

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 addition to improved control of secondary water chemistry, 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.

Tube-To-Tube Wear

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, the 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 was determined to be a result of a combination of factors including: nonconservatism in the margin to tube buckling in the design, the outside temperature of the steam generator shell is cooler than the value used in the design, the tube preload is less tensile than the value used in the design analysis, and the lateral loads or accelerations are sufficient to cause the observed wear. Since this degradation is readily managed through the licensees' Steam Generator Tube Integrity Programs, no additional regulatory action was deemed necessary. Additional information can be found in the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. ML13178A358.

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 stated that the U-bend portion of some of the tubes experienced fluid elastic instability in the inplane 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.

Ultimately, the plant did not restart. By letter dated June 12, 2013, Southern California Edison 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.

The NRC staff assessed the lessons learned from the San Onofre event. The results of that effort were documented on March 6, 2015. Additional information can be found under ADAMS Accession No. ML15062A125.

<u>Low Alloy Steel Channel Head Corrosion Operating Experience</u>

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. On October 3, 2013, the NRC staff issued IN 2013-20, “Steam Generator Channel Head and Tubesheet Degradation,” to address this issue. As of February 2016, 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 has been observed at a couple U.S. facilities. The U.S. industry’s response has been effective at addressing this issue; therefore, no further regulatory action has been deemed necessary.

1.3.2 Current Safety and Regulatory Issues

The NRC and its licensees are evaluating and resolving the following potential safety and regulatory issues:

- baffle-former bolts
- digital instrumentation and control systems
- open phase conditions in electric power system
- risk-informing regulations and processes
- SFP neutron-absorbing materials
- staff readiness to transition plants from construction to operations
- staff readiness to transition plants from operation to decommissioning
- subsequent license renewal
- Project Aim

Baffle-Former Bolts

The core baffle is a portion of the reactor vessel internals in a Westinghouse PWR. The core baffle is located within the core barrel and functions to direct the coolant flow through the core and provide some lateral support to the fuel assemblies. Vertical baffle plates are bolted to the edges of horizontal former plates, which are attached to the inside surface of the core barrel. There are typically eight levels of former plates located at various elevations within the core barrel. The bolts that secure the baffle plates to the former plates are referred to as baffle-former bolts. To cool the baffle structure, some water flowing through the reactor vessel is directed between the core barrel and the baffle plates in either a downward direction (i.e., downflow configuration), or an upward direction (i.e., upflow configuration).

Degradation of the baffle-former bolts was first noted in the late 1980s in PWR facilities outside the United States. The NRC communicated operating experience on baffle-former bolt degradation to U.S. licensees in IN 98-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants,” dated March 25, 1998. On January 9, 2012, EPRI issued MRP-227-A, “Power Reactor Internals Inspection and Evaluation Guidelines,” which includes an inspection of the baffle-former bolts during the timeframe when bolt degradation is most likely to appear, as demonstrated by operating experience. The NRC endorsed MRP-227-A in 2012. The guidelines of MRP-227-A provide for the development of an aging management program for PWR reactor vessel internals that meets the NRC requirements for issuance of a renewed operating license.

The degradation of the baffle-former bolts is attributed to irradiation assisted stress corrosion cracking. Baffle-former bolts are subjected to significant stresses and irradiation over years of plant operation. PWRs with a downflow configuration place additional stress on the baffle-former bolts due to the pressure differential across the vertical baffle plates. At this time, significant degradation of the baffle-former bolts has only been observed in Westinghouse four-loop PWR reactors with the downflow configuration and bolts made of type 347 stainless steel. Seven reactors in the U.S. match this description.

In spring 2016, two U.S. nuclear power plants identified a large number of type 347 stainless steel baffle-former bolts with indications of degradation during the performance of ultrasonic inspections following MRP-227-A guidelines. In general, both units replaced potentially degraded bolts with type 316 stainless steel bolts in an improved design. The NRC has inspected repair activities at these facilities.

The NRC performed a risk-informed evaluation of the safety impact that degradation of baffle-former bolts could present to operating reactors. The NRC concluded that this issue did not pose an immediate shutdown of any facilities. Analyses will be performed to determine the material condition of the baffle-former bolts that were removed from the two units and provide additional insights on the degradation mechanism. The NRC is considering a future generic communication to address the issue, which will be informed by the results of these ongoing analyses. The remaining susceptible reactors have indicated that they will accelerate schedules for performing the MRP-227-A inspection of the baffle-former bolts. The U.S. nuclear industry has formed a working group to consider potential changes to the MRP-227-A inspection regime. The NRC is monitoring the industry response to this issue and will take appropriate regulatory actions.

Digital Instrumentation and Control Systems

The NRC maintains a robust regulatory program for ensuring the safety and security of nuclear facilities protected and operated with analog and digital instrumentation and control systems. Using its current regulatory infrastructure, the NRC staff continues to review and approve license amendments for specific digital instrumentation and control systems, and evaluate new reactor applications that fully incorporate highly integrated digital technologies. However, the efficiency and predictability of the processes can be improved. The staff is developing an action plan to implement an integrated strategy to modernize the NRC's digital instrumentation and control regulatory infrastructure and provide for consistent, predictable, and efficient implementation of digital technology.

A Steering Committee, comprised of NRC senior managers, is providing oversight in the development of the integrated action plan. The Steering Committee provides oversight for updating and implementing the plan in order to ensure a sound strategy to modernize the NRC's digital instrumentation and control regulatory infrastructure. The plan is being developed through engagement of Institute of Electrical and Electronics Engineers (IEEE) standards setting committees, digital instrumentation and control vendors, licensees, and other external stakeholders in order to reach a common understanding of the digital instrumentation and control regulatory challenges, priorities and potential solutions. The plan is intended to encompass all NRC digital instrumentation and control activities and regulatory challenges. Examples include incorporation of IEEE standards into NRC regulations, updates to NRC's policy on common-cause failure, and guidance for applicants and licensees to implement digital instrumentation and control systems. The plan will ensure that any new or revised requirements: (1) are performance-based (rather than prescriptive), (2) are technology neutral, (3) apply in the same manner to operating and new reactors, and (4) do not pose an unnecessary impediment to advancement in nuclear applications of digital technology. Also, the integrated action plan will ensure that the modernized regulatory infrastructure will improve the predictability and consistency of the agency's regulatory process for licensing and oversight for digital instrumentation and control systems.

In May 2016, the NRC staff submitted the initial integrated action plan to the Commission for review and approval.

Open Phase Conditions in Electric Power Systems

Operating experience has identified design vulnerabilities associated with open phase conditions in offsite power systems at operating nuclear plants. These events involved offsite power supply circuits that were rendered inoperable by open-circuited phase conditions. This condition can degrade the performance capabilities of both offsite and onsite power systems. The operating events indicated that the design of the electric power systems to minimize the probability of losing electric power from any of the remaining power supplies as a result of, or coincident with, the loss of power from the transmission network were inadequate because it did not take into account the possibility of the loss of a single phase between the transmission network and the onsite power distribution system.

The January 30, 2012, operating event at Byron Station, Unit 2, revealed a significant design vulnerability, which resulted in the loss of safety functions for electric power systems. At Byron Station, Unit 2, both offsite and onsite electric power systems were unable to perform their intended safety functions to provide electric power to the engineered safety feature buses with sufficient capacity and capability to permit functioning of SSCs important to safety. The staff determined that a design-basis event concurrent with an undetected open phase condition would likely have resulted in the plant exceeding criteria specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and the accident analyses assumptions. Based on the Byron Station operating event, the staff issued IN 2012-03, "Design Vulnerability in Electric Power System," dated March 1, 2012.

A review of other operating experience identified similar design vulnerabilities associated with single phase open circuit conditions at South Texas, Unit 2, Beaver Valley Power Station, Unit 1, and a single event that affected Nine Mile Point, Unit 1, and the neighboring James A. FitzPatrick power plant.[5] These events involved offsite power circuits that were rendered inoperable due to an open circuit in one phase. In each instance (except in the South Texas,

5 South Texas Project, Unit 2, Licensee Event Report (LER) 2001-001 (ADAMS Accession No. ML011010017); Beaver Valley Power Station, Unit 1, LER 2007-002-00 (ADAMS Accession No. ML080280592); Nine Mile Point Nuclear Station, Unit 1, LER 2005-04 (ADAMS Accession No. ML060620519); and James A. FitzPatrick LER 2005-006 (ADAMS Accession No. ML060610079).

Unit 2, event), the condition went undetected for several weeks because offsite power was not aligned to the engineered safety feature buses during normal operation and the surveillance procedures, which recorded phase-to-phase voltage, did not identify the loss of the single phase. At South Texas, Unit 2, offsite power was normally aligned to the engineered safety feature and nonsafety plant buses, and the reactor was manually shut down by the operator when the three circulating water pumps were tripped by the open phase condition. In addition, operating experience has identified three similar international events with one or two open phase conditions at reactors located in Canada, Sweden, and United Kingdom.

In the domestic and international events discussed above, the protective relaying schemes did not detect the open phase conditions due to inadequate detection schemes. The open phase event at Byron Station[6], Unit 2, resulted in degraded and unbalanced voltage conditions on redundant engineered safety features buses, which led to the tripping of equipment required for normal plant operations and safe shutdown. The inability of the protection scheme to detect an open phase condition and automatically transfer power from the affected electric power system allowed the degraded offsite power system to remain connected to engineered safety features buses, and prevented other onsite AC sources (e.g., emergency diesel generators) from starting and powering these buses. As a result, certain equipment required for safe operations remained powered by the degraded AC source and were put in jeopardy to either rely on internal safety features to lockout and protect the vulnerable components or risk damage from overheating. Furthermore, equipment relied on for safe shutdown was also at risk of being unavailable for a period of time outside the plant's accident analysis, even after restoration of an operable power source.

On July 27, 2012, the staff issued Bulletin 2012-01, "Design Vulnerability in Electric Power System," to confirm that licensees comply with 10 CFR 50.55a(h)(2), 10 CFR 50.55a(h)(3), General Design Criterion 17, "Electric Power Systems," or applicable principal design criteria specified in the updated final safety analysis report. The NRC staff has reviewed the information that the licensees provided and the details of this review are documented in Bulletin 2012-01, "Design Vulnerability in Electric Power System: Summary Report," dated February 26, 2013. The staff concluded that this design vulnerability exists at all operating plants, except for Seabrook Nuclear Power Plant because of plant-specific switchyard features.

The staff developed Branch Technical Position 8-9, "Open Phase Conditions in Electric Power System," in July 2015, to provide guidance to staff on reviewing any licensing actions to address this design vulnerability. The licensees have implemented interim corrective actions and compensatory measures to address the operability of the electric power system until plant modifications are completed by December 31, 2018. Plant modifications can include installing a system or component that will detect and automatically isolate a single-phase open circuit condition and/or a high impedance ground fault condition on the electrical systems at applicable nuclear power plant sites.

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6 Byron Station, Units 1 and 2, LER 12-001-00 (ADAMS Accession No. ML12090A492); and Byron Station, Units 1 and 2, LER 12-001-01 (ADAMS Accession No. ML12272A358).

Risk-Informing Regulations and Processes

The NRC is advancing the use of risk-information in regulatory decisionmaking and processes, while continuing to emphasize defense-in-depth and safety margins. Several related initiatives, such as the NRC's Risk Informed Steering Committee, and risk-informed improvements to standard technical specifications serve as key examples that are shaping the direction that the agency will take in regard to risk-informed decisionmaking.

NRC's Risk Informed Steering Committee

The NRC's Risk Informed Steering Committee is an NRC senior management committee that provides strategic direction to the NRC staff to advance the use of risk-informed decisionmaking in licensing, oversight, rulemaking, and other regulatory areas, consistent with the Commission's "Policy Statement on Use of PRA Methods in Nuclear Activities," dated August 16, 1995. The Committee is chaired by the Director of the Office of Nuclear Reactor Regulation, with membership consisting of Deputy Office Directors from the Offices of New Reactors, Nuclear Regulatory Research, Nuclear Materials Safety and Safeguards, Nuclear Security and Incident Response, and Nuclear Reactor Regulation, as well as the Region I Administrator.

The nuclear industry has its own Risk Informed Steering Committee, which is a counterpart to the NRC's Committee, with its membership comprising licensee chief nuclear officers and other senior level executives, as well as representation from the Nuclear Energy Institute. The NRC Risk Informed Steering Committee has held several public meetings with the industry's committee. The NRC and industry each agreed to form two working groups that focused on guidance in two selected areas related to probabilistic risk assessment (PRA) technical adequacy and dealing with uncertainties in risk-informed decisionmaking. These working groups formed problem statements and action plans that have been provided to the applicable line organizations for consideration and continued work.

The NRC Risk Informed Steering Committee is also providing direction to the NRC staff concerning efforts to provide credit for mitigating strategies put in place in response to the Commission Orders after the events at the Fukushima Dai-ichi nuclear power plant. The NRC's Risk Informed Steering Committee will continue to hold public meetings with the industry to discuss current and upcoming risk-informed initiatives of interest.

Risk-Informed Improvements to Standard Technical Specifications

In 1992, the NRC issued the improved Standard Technical Specifications to clarify the content and form of requirements necessary to ensure safe operation of nuclear power plants in accordance with 10 CFR 50.36, "Technical Specifications." As the Standard Technical Specifications mature, areas for improvement have been identified. One process used to efficiently initiate changes to the Standard Technical Specifications involves the industry-sponsored Technical Specifications Task Force submitting a proposed change, commonly known as a "Traveler," to the NRC for review, approval, and subsequent incorporation into the next revision of the Standard Technical Specifications.

The NRC reviews the proposed change, with the end product being a model application, a model safety evaluation, and a review plan that licensees may use in subsequent license amendment requests. Licensees applying to incorporate these proposed changes into their

Technical Specifications must provide a plant-specific justification acceptable to the NRC staff in their amendment request. The NRC staff is currently reviewing several risk-informed license amendment requests in accordance with Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," and Traveler TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times (CTs) – RITSTF Initiative 4b." TSTF-425/RITSTF Initiative 5b provides a risk-informed methodology to identify, assess, implement, and monitor proposed changes to frequencies of technical specification surveillance requirements. TSTF-505/RITSTF Initiative 4b allows licensees to modify selected required actions to permit extending completion times, provided risk is assessed and managed within an acceptable configuration risk management program. These initiatives are intended to maintain and improve safety through incorporation of risk assessment and management techniques in the Technical Specifications, while reducing unnecessary burden. The staff continues to work on the risk-informed technical specifications initiatives to add a risk-informed component to the Standard Technical Specifications.

Spent Fuel Pool Neutron-Absorbing Materials

The NRC requires that power reactor license holders maintain SFP subcriticality in accordance with 10 CFR 50.68, "Criticality Accident Requirements," General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," and other equivalent regulatory criteria. The NRC has a similar requirement included in the technical specifications for nonpower reactors.

Neutron-absorbing materials have been used in SFPs for more than 30 years to allow for increases in SFP storage capacity while maintaining safety margins against inadvertent criticality. Plates or sheets of these materials are used in SFP racks and are comprised of a compound, alloy, or a composite material that serves as a matrix to contain a neutron absorber nuclide, primarily boron-10. Several types have been deployed and include the following:

- boron carbide (B_4C) in a silicone polymer
- B_4C in a phenol formaldehyde resin matrix
- B_4C in an aluminum matrix with aluminum cladding
- natural boron in a stainless steel matrix
- B_4C in an aluminum metal matrix composite

The NRC is performing confirmatory research on SFP neutron-absorbing materials used in the U.S. commercial nuclear power industry, the surveillance methodologies used by the industry, and the surveillance intervals of the neutron-absorbing materials used in the SFPs.

Furthermore, the NRC has issued three technical letter reports[7] discussing some of the methods that license holders use to monitor the degradation of neutron-absorbing materials, the uncertainties in the methodologies employed to monitor the performance, and the degradation mechanisms.

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7 "Boraflex, RACKLIFE and BADGER: Description and Uncertainties," dated September 30, 2012, "Initial Assessment of Uncertainties Associated with BADGER Methodology," dated September 30, 2012, and "Monitoring Degradation of Phenolic Resin-Based Neutron Absorbers in Spent Nuclear Fuel Pools," dated June 5, 2013.

Operating experience includes several instances of degradation and deformation of neutron-absorbing materials in SFPs, as described in IN 09-26, “Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool,” dated October 28, 2009, and other earlier communications. Although there have been no criticality incidents, licensee performance with regard to detection and mitigation of neutron-absorbing materials degradation has resulted in several licensees being outside their licensing bases. This experience has raised concerns with the adequacy of licensees’ current SFP neutron-absorbing materials monitoring.

Therefore, the NRC issued a draft GL, “Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools,” on March 11, 2014, to request licensees to provide information that demonstrates that credited neutron-absorbing materials in the SFP of power reactors and the fuel storage pool, reactor pool, or other wet locations designed for the purpose of fuel storage in nonpower reactors are in compliance with the licensing and design basis, and applicable regulatory requirements. The staff issued the final document as GL 2016-01, “Monitoring of Neutron-Absorbing Materials In Spent Fuel Pools,” on April 7, 2016. The NRC staff will review licensees responses and determine wither further regulatory action is needed.

Staff Readiness to Transition Plants from Construction to Operations

With respect to the NRC readiness to transition new reactors from construction to operations, an NRC staff-level transition working group was established in 2013 to develop an integrated plan that identified all regulatory functions necessary to support the transition of the Vogtle and V.C. Summer AP1000 units from construction to operation. In September 2014, NRC staff summarized the working group’s results in the report, “Assessment of the Staff’s Readiness to Transition Regulatory Oversight and Licensing as New Reactors Proceed from Construction to Operation.” The report included 21 readiness issues with associated options and recommendations, with an emphasis on ensuring continuity during the oversight transition. The resolution of these recommendations is currently being tracked and reported to management on a regular basis.

In response to one of the readiness issues, NRC staff is drafting a detailed implementation plan to transfer regulatory oversight and licensing for the AP1000 design center from the New Reactor Licensing and Construction organizations implementing the Construction Reactor Oversight Program (cROP) framework to the appropriate operations organizations responsible for the Reactor Oversight Process. The draft implementation plan is currently undergoing internal review and approval, and will be distributed to public stakeholders for comments and discussion at a future Reactor Oversight Process public meeting. Once comments have been received and addressed, a final version will be forwarded for approval by the impacted Office Directors and Region II Administrator. The plan is intended to be a living document and will be revised, as necessary. In line with the NRC’s Principles of Good Regulation, the readiness plan will be executed with openness and clarity by coordinating with the licensees and communicating with the public, as appropriate.

Staff Readiness to Transition Plants from Operation to Decommissioning

When a utility decides to close a nuclear power plant permanently, the facility must be decommissioned by safely removing it from service and reducing residual radioactivity to a level that permits release of the property and termination of the license. The NRC has strict rules governing nuclear power plant decommissioning, involving cleanup of radioactively contaminated plant systems and structures, and removal of the radioactive fuel. These requirements protect workers and the public during the entire decommissioning process, and the public after the license is terminated.

The requirements for decommissioning a nuclear power plant are set out in several NRC regulations.[8] In August 1996, revised rules went into effect that redefined the decommissioning process. The revised regulation required licensees to provide the NRC with early notification of planned decommissioning activities. The rules do not allow major decommissioning activities to be undertaken until after certain information has been provided to the NRC and the public.

Most regulations in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," as well as other parts of NRC's regulations were intended to be applicable only to a reactor that is authorized to operate. Once a licensee has submitted certifications of permanent cessation of operation and permanent removal of fuel from the reactor vessel, in accordance with 10 CFR 50.82, "Termination of License," it is no longer authorized to operate its nuclear power plant. Because the risks at a permanently shut down reactor are significantly lower compared to the risks from operating reactors, certain reactor regulations and specific license conditions may not be applicable to permanently shutdown reactors. In these circumstances, the licensee may apply for an exemption to the regulation or amendment to its license, and the NRC staff will review and evaluate the licensee's specific request.

In 2013 and 2014, five power reactor units permanently ceased operation (i.e., Kewaunee; Crystal River, Unit 3; San Onofre, Units 2 and 3; and Vermont Yankee). These were the first reactors to permanently cease operations since 1998 - a span of nearly 15 years without a power reactor permanently shutting down. In addition, the licensees for seven other reactor units have announced an intent to permanently cease operations in the next few years. These reactor units include: Fort Calhoun Station; James A. FitzPatrick Nuclear Power Plant; Clinton Power Station, Unit 1; Quad Cities Nuclear Power Station, Units 1 and 2; Pilgrim Nuclear Power Station; and the Oyster Creek Nuclear Generating Station. All of the recent reactors transitioning to decommissioning have requested amendments to their licenses and exemptions from the NRC's regulations. Most changes involve modifications to staffing, security, emergency preparedness and financial assurance requirements; and the deletion of license conditions and technical specifications that are no longer applicable. The NRC staff reviews each request with a focus on safety and the individual circumstances at each site and whether the request, if approved, would maintain an adequate level of protection. To the maximum extent practicable, the staff has used precedent from previously decommissioned plant evaluations as a basis for the current licensing action reviews.

Since the last plants that entered into decommissioning in the 1990s, there have been many new and revised regulatory requirements for operating plants related to emergency preparedness and security rules and Orders that address lessons learned from the terrorists attacks of September 11, 2001, and the 2011 Fukushima Dai-ichi nuclear accident in Japan. These requirements also need to be considered for their applicability in decommissioned plants.

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8 Title 10 of the *Code of Federal Regulations*, Part 20 Subpart E, and Part 50, Sections 50.75, 50.82, 51.53, and 51.95.

As licensing actions were issued, starting with the shutdown of Kewaunee in 2013, decommissioning experience has been acquired, and significant efficiencies for processing subsequent decommissioning reviews have been achieved.

In SRM-SECY-14-0118, “Request By Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements,” dated December 30, 2014, the Commission directed the staff to proceed with a rulemaking on decommissioning. The Commission stated that the rulemaking should address issues such as:

- the graded approach to emergency preparedness
- lessons learned from the plants that have already (or are currently) going through the decommissioning process
- the advisability of requiring a licensee's Post-Shutdown Decommissioning Activity Report to be approved by NRC
- the appropriateness of maintaining the three existing options for decommissioning and the timeframes associated with those options
- the appropriate role of state and local governments and non-governmental stakeholders in the decommissioning process
- other issues deemed relevant by the NRC staff

As documented in SECY-15-0014, “Anticipated Schedule and Estimated Resources for a Power Reactor Decommissioning Rulemaking,” dated January 30, 2015, the staff has initiated the decommissioning rulemaking process. One objective of this potential rulemaking would be to provide a more efficient and predictable decommissioning transition process. In addition, the potential rulemaking would support the Principles of Good Regulation, including openness, clarity, and reliability. An Advance Notice of Proposed Rulemaking was published in November 2015. The NRC received 161 public comment submissions, which are being considered as part of the development of the regulatory basis for the proposed rule. Subsequently, the staff will submit a proposed rule for Commission review and approval. Should the Commission decide to proceed with this rulemaking, the staff will not provide the draft final rule to the Commission for approval until calendar year 2019 or beyond, depending on the competing demands on agency decommissioning experts.

Subsequent License Renewal

The NRC’s current regulatory framework in 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” supports the receipt and review of a subsequent license renewal application. Specifically, 10 CFR 54.31(d) states that “a renewed license may be subsequently renewed in accordance with all applicable requirements.”

Several industry representatives have expressed an interest in operating plants beyond 60 years. In SRM-SECY-14-0016, “Ongoing Staff Activities to Assess Regulatory Considerations for Power Reactor Subsequent License Renewal,” dated August 29, 2014, the Commission concluded that the current regulatory framework for the first license renewal was sound and sufficient to provide reasonable assurance that the power reactors can safety operate beyond 60 years. On November 5, 2015, the Commission received a formal letter of intent to pursue such a renewal for Surry Power Station, Units 1 and 2, in 2019. On June 7, 2016, the Commission received a formal letter of intent to pursue a second license renewal for Peach Bottom Atomic Power Station, Units 2 and 3, in 2018.

To support the review of the application, the NRC staff has developed guidance documents to address the unique aging management needs for a subsequent license renewal. The unique aging management needs include technical issues such as reactor pressure vessel neutron embrittlement at high fluence, irradiation assisted stress corrosion cracking of reactor internals and primary system components, concrete degradation and electrical cable qualification and condition assessment. These guidance documents were developed by making the necessary revisions to the existing license renewal guidance documents, which include NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 2, both issued in December 2010. The NRC staff plans to issue the updated guidance by mid-2017. Section 14.1.4.3 of this report describes the subsequent license renewal activities in more detail.

Project Aim

Project Aim is an NRC initiative to improve agency efficiency, effectiveness, and agility, driven largely by the need to align agency regulatory work environment, structure, and processes with numerous fact-of-life changes. The agency has grown significantly to enhance security and incident response (subsequent to the September 11, 2001, terrorist attacks) and to prepare for projected growth in the use of nuclear power in the U.S. The forecasted growth did not occur because of market conditions in the nuclear industry that resulted in fewer new nuclear facilities and the early closure of existing plants.

The NRC's Executive Director for Operations established Project Aim in coordination with the Chief Financial Officer in June 2014 to enhance the agency's ability to plan and execute its mission while adapting in a timely and effective manner to a dynamic environment. The project team gathered perspectives from internal and external stakeholders to forecast the future workload and operating environment. Based on analysis of these perspectives, literature review, and the evaluation of the NRC's current state, the team and NRC senior leadership identified key strategies to transform the agency to improve the effectiveness, efficiency, and agility of the NRC. If implemented, these strategies will better position the NRC to respond to new safety and regulatory challenges without compromising our important mission and without affecting our ability to demonstrate organizational values and Principles of Good Regulation.

In late January 2015, NRC staff provided a number of recommendations to the Commission. The staff's report proposed that the NRC could function more efficiently by performing the following:

- right-sizing the agency while retaining appropriate skill sets needed to accomplish its mission

- streamlining agency processes to use resources more wisely

- improving timeliness in regulatory decisionmaking and responding quickly to changing conditions

- promoting unity of purpose with clearer agencywide priorities

On June 8, 2015, the Commission approved many of the recommendations presented by the staff's report. A wide range of implementation activities are currently underway and are being tracked as 19 discrete tasks. Among the Project Aim activities, the Commission directed the staff to undertake a "re-baselining" effort to identify work that can be shed or eliminated, deferred, or done with fewer resources. This task, which involved a broad review of the agency's workload, resulted in recommended efficiencies for Commission consideration as documented in SECY-15-0009, "Recommendations Resulting from the Integrated Prioritization and Re-Baselining of Agency Activities," dated January 31, 2016. The Commission approved almost all of those recommendations in SRM-SECY-16-0009, which was issued on April 13, 2016. The staff began developing implementation plans to achieve these approved efficiencies in an open, collaborative, and transparent manner.

1.3.3 Major Regulatory Accomplishments

Since its previous U.S. National Report was issued in 2013, the NRC has achieved numerous regulatory accomplishments. The following are some of the major items.

- continued storage
- issuance of new licenses
- Fukushima lessons learned
- revisions to the petition for rulemaking process
- *Southern Exposure 2015* exercise

Continued Storage

On August 31, 1984, the Commission issued the Waste Confidence Decision and Rule (49 FR 34658 and 49 FR 34688), codified in 10 CFR 51.23, "Environmental Impacts of Continued Storage of Spent Nuclear Fuel Beyond the Licensed Life for Operation of a Reactor." This regulation represented the Commission's generic environmental determination that spent fuel can be stored safely and without significant environmental impacts for a period of time after the end of the licensed life for operation. This generic determination meant that the NRC did not need to consider the storage of spent fuel after the end of a reactor's licensed life for operation in documents that support its reactor and spent fuel storage application reviews in accordance with the National Environmental Policy Act (NEPA). The Commission conducted subsequent reviews of the decision and rule in 1990 and 1999 (55 FR 38474; September 18, 1990 and 64 FR 68005; December 6, 1999).

On December 23, 2010, the Commission published in the *Federal Register* a revision of the Waste Confidence Decision and Rule to reflect information gained from experience in the storage of spent fuel and the increased uncertainty in the siting and construction of a permanent geologic repository for the disposal of spent nuclear fuel and high-level waste (75 FR 81032 and 75 FR 81037). Several parties challenged the Waste Confidence Decision, which provided the regulatory basis for the rule. On June 8, 2012, the United States Court of Appeals for the District of Columbia Circuit vacated the NRC's Waste Confidence Decision and Rule because it did not comply with NEPA. The Court concluded that to comply with NEPA, the agency was required to identify the impacts of failing to develop a repository, and that its analyses of spent fuel pool leaks and fires provided in the Waste Confidence Decision could not support a 'finding of no significant impact' (because the pool leak analysis was not sufficiently forward-looking and the analysis of fires addressed only the likelihood, but not the consequence, of a fire).

In response to the court's ruling, the Commission directed the NRC staff to proceed with a rulemaking, supported by an environmental impact statement. On August 26, 2014, the Commission approved the revised 10 CFR 51.23 rule and associated NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel." The revised rule and NUREG were published on September 19, 2014. The revised rule adopts the generic impact determinations made in NUREG-2157 and codifies the NRC's generic determinations regarding the environmental impacts of continued storage of spent nuclear fuel beyond the term of a reactor's operating license (i.e., those impacts that could occur as a result of the storage of spent nuclear fuel at at-reactor or away-from-reactor sites after a reactor's licensed life for operation and until a permanent repository becomes available). Additionally, the NRC adopted the new name, "Continued Storage," as a replacement for the long-standing historical name, "Waste Confidence." This name change reflects the fact that the analysis considers the storage of spent fuel under three separate scenarios: (1) the short-term timeframe considers 60 years beyond the reactor's license term (including two renewal terms); (2) the long-term timeframe considers an additional 100 years; and (3) the indefinite timeframe assumes that no repository becomes available.

In 2014, the parties who challenged the 2010 Waste Confidence Update filed new petitions for review of the Continued Storage final rule in the District of Columbia Circuit, arguing that that the generic environmental impact statement insufficiently addressed several issues under NEPA. In a decision issued on June 3, 2016, in *New York v. NRC*, the Court denied the petitions and upheld the NRC's Continued Storage final rule.

Issuance of New Licenses

Enrico Fermi Nuclear Plant, Unit 3

On May 1, 2015, the NRC issued a combined license to DTE Energy Company for the Enrico Fermi Nuclear Plant, Unit 3. The license authorizes DTE Energy Company to build and operate (with specified conditions) a General Electric-Hitachi's Economic Simplified Boiling Water Reactor at the Fermi site, adjacent to the company's existing reactor near Newport, Michigan. This is the fifth combined license issued by the NRC and the first combined license issued for an Economic Simplified Boiling Water Reactor. The combined license is valid for 40 years from the date of the Commission finding that the acceptance criteria in the combined license are met.

Watts Bar Nuclear Plant, Unit 2

On October 22, 2015, the NRC issued a full power facility operating license for Watts Bar Nuclear Plant, Unit 2, to the Tennessee Valley Authority. The operating license is valid for 40 years. The Watts Bar Nuclear Plant, Unit 2, is the first initial operating license granted by the NRC under 10 CFR Part 50 since the operating license was issued for Watts Bar Nuclear Plant, Unit 1, on February 7, 1996. Additional information about the licensing of Watts Bar Nuclear Plant, Unit 2, can be found in Section 18.1.2 of this report.

South Texas Project, Units 3 and 4

On February 12, 2016, the NRC issued combined licenses to Nuclear Innovation North America LLC for the South Texas Project, Units 3 and 4. The licenses authorize Nuclear Innovation North America LLC to construct and STP Nuclear Operating Company to operate (with specified conditions) two Advanced Boiling Water Reactors at the South Texas Project site, adjacent to the two existing reactors in Matagorda County, Texas. These are the sixth and seventh combined licenses issued by the NRC, and the first combined licenses issued for an Advanced Boiling Water Reactor. 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.

Fukushima Lessons Learned

The NRC has implemented a number of safety enhancements based on lessons learned from the accident at the Fukushima Dai-ichi nuclear plant in Japan. As part of its activities in this area, the NRC has interacted extensively 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 the accident.

Orders, Rulemaking, and Seismic and Flooding Evaluations

The NRC continues to review nuclear power plant licensees' plans to achieve compliance with the mitigation strategies and SFP instrumentation orders, which were issued in March 2012. The NRC has issued both interim and final staff evaluations and is in the process of inspecting the licensees' implementation of these safety improvements. On October 4, 2014, the first licensee informed the NRC that one of its nuclear plants was fully compliant with both orders. All plants are expected to be in compliance with the SFP order by the end of December 2016. Nearly all plants will be in compliance with the mitigating strategies order by the end of 2016, with the exception of those plants whose mitigating strategies also rely on implementing the hardened vent order (in those cases, all other enhancements will be in place at that time). Nuclear power plant licensees are also reevaluating seismic and flooding hazards for their sites and are performing evaluations to ensure that mitigating strategies are available to address potentially more severe external events. The need for mitigating strategies to address beyond-design-basis external events is included in a rule that will be provided to the Commission for approval by the end of 2016, and the majority of licensees are expected to complete the evaluations and any necessary modifications in 2017.

In June 2014, NRC staff received the licensees' integrated plans for compliance with Phase 1 (wetwell venting) of the revised severe-accident-capable hardened vents order, which was first issued in March 2012 and subsequently revised in June 2013. The staff has issued interim staff evaluations of those plans. Licensees submitted their plans for Phase 2 (drywell venting) of the revised order in December 2015, and are required to fully implement these plans on a staggered basis and by no later than June 2019.

The NRC requested that nuclear power plant licensees reevaluate potential seismic and flooding hazards. If these reevaluated hazards are not bound by the current design basis, the licensee will determine whether interim protection measures are needed while a longer-term evaluation is completed.

The NRC staff is reviewing the flooding hazard reevaluations and is issuing assessments, as described in COMSECY-15-0019, “Closure Plan for the Reevaluation of Flooding Hazards for Operating Nuclear Power Plants,” dated June 30, 2015. Under this plan, licensees will perform an assessment of the mitigating strategies to ensure that they can address the reevaluated hazards. Licensees are expected to complete the mitigation strategies assessment to the reevaluated flood hazard by December 2016. The plan also includes provisions for additional evaluations, beyond those associated with mitigating strategies, depending on the reevaluated hazard.

The NRC has received the seismic hazard reevaluation reports from its licensees and has determined the final list of operating reactor sites where the development of a seismic risk evaluation is warranted. Approximately one third of the sites are expected to perform a seismic PRA, the first of which will be due to the NRC in March 2017. Some sites will perform limited-scope evaluations (i.e., a high-frequency evaluation, low-frequency evaluation, or SFP evaluation). Nine sites have screened out of any further evaluation.

The NRC also requested information from licensees on emergency preparedness staffing and communications, which is further discussed in Section 16.9 of this report.

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 in November 2012. Identified discrepancies were entered into the respective licensee’s corrective action program.

Rulemaking activities related to the requirements of the orders and other NTTF recommendations are also proceeding as scheduled. As mentioned above, the NRC has received comments from external stakeholders on the proposed Mitigation of Beyond-Design-Basis Events rulemaking and plans to provide a final rule to the Commission by the end of 2016. With regard to the planned second post-Fukushima rulemaking, the “Containment Protection and Release Reduction” rulemaking, the Commission directed the staff not to proceed with that rulemaking after determining that the safety benefit is already being achieved through the hardened containment vent order and additional requirements in that area would not be justified.

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. Because sufficient resource flexibility, including availability of critical skill sets, did not exist at the time that the staff prioritized the recommendations, the staff prioritized the other external hazards evaluation as a Tier 2 activity and are working to resolve the issue as part of the completion of Fukushima-related activities.

The NRC is moving forward with resolving the remaining open Tier 2 and 3 recommendations that have not already been addressed through, for example, the ongoing rulemaking initiative or other Tier 1 activities. On October 29, 2015, the NRC staff delivered a paper, SECY-15-0137, to the Commission that described the resolution paths for these recommendations. The NRC staff recommended that some could be closed now, while others would benefit from either further stakeholder interaction or additional analysis or documentation. With respect to its evaluation of external hazards other than seismic and flooding, SECY-15-0137 discusses the staff's plans to further develop a process to determine if there is a need and justification to impose a new regulatory requirement under the backfit rule or if the NRC should take some other regulatory action (e.g., issue a generic communication) to enhance protection from other hazards. The proposed process would entail the NRC applying screening criteria to determine if a hazard could be screened out generically and, if not, to take into account site- or hazard-specific information. SRM-SECY-15-0137, dated February 8, 2016, approved the staff's resolution plans for these Tier 2 and 3 recommendations. The Commission approved closing a subset of these recommendations and directed the staff to continue issuing status updates on those recommendations that remain open. The staff provided the Commission with final assessments concluding that no additional regulatory actions were warranted for containments other than BWR Mark I and II designs, hydrogen control, or enhanced instrumentation. The assessments of the remaining Tier 2 and 3 recommendations are expected to be completed by late 2016.

Spent Fuel Storage

As part of the resolution of a recommendation to assess the merits of expediting the transfer of spent fuel from pools to dry storage casks, the NRC completed an SFP consequence study, documented in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014. The study considered an elevated SFP (similar to the one at Fukushima and 23 U.S. reactors) undergoing a postulated earthquake with ground motion stronger than the maximum earthquake reasonably expected to occur for the reference plant. The study examined both a full SFP and one with less fuel, after removing older 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 through a draindown. The study also showed that even if this particular pool was damaged, the fuel could be kept cool by air circulation, in all but a few analyzed conditions.

In cases where the analysis led to fuel damage, the study predicted that, with existing emergency procedures, no offsite early fatalities attributable to acute radiation exposure would occur. Those emergency measures could involve 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 containing a smaller inventory of radioactive fission products) 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 used the study to inform regulatory evaluations in COMSECY-13-0030, which supported a recommendation to the Commission to close the issue. The Commission agreed with the staff's recommendation that this Tier 3 activity be closed and that no further generic assessments be pursued related to possible regulatory actions to require the expedited transfer of spent fuel to dry cask storage.

Public Interaction

Since 2012, the NRC has conducted hundreds of public meetings related to the lessons learned from the Fukushima accident. Many NRC representatives have presented information on Fukushima and the lessons learned to university scholars and professional scientific and engineering societies. The NRC's annual Regulatory Information Conferences have also included presentations on activities associated with Fukushima lessons learned. These meetings, interactions, and exchanges of information have been instrumental in obtaining input from stakeholders, which has been and will continue to be factored into the NRC's regulatory activities and lessons learned implementation plans.

Revisions to the Petition for Rulemaking Process

In the U.S., federal law allows individuals, companies, states, local governments, and federally-recognized Tribes to ask agencies to adopt, revise, or withdraw existing regulations. This type of request is known as a petition for rulemaking. On October 7, 2015, the NRC published a final rule amending its petition for rulemaking regulations in 10 CFR Part 2, "Agency Rules of Practice and Procedure." The final rule makes the petition for rulemaking process more efficient and effective, while enhancing transparency and public participation in the process. Specifically, the final rule:

- expands a petitioner's access to the NRC by allowing consultation with NRC staff both before and after filing a petition for rulemaking
- restructures and clarifies the content requirements for a petition for rulemaking
- clarifies the NRC's petition for rulemaking evaluation criteria
- explains the NRC's internal process for receiving, closing, and resolving a petition
- updates information for tracking the status of petitions and subsequent rulemaking actions

A Web site describing the NRC's petition for rulemaking process is available on the NRC's public Web page at http://www.nrc.gov/about-nrc/regulatory/rulemaking/petition-rule.html. The rulemaking process was open and transparent and members of the public were given the opportunity to comment on the proposed rule.

Southern Exposure 2015 "Whole Community" Emergency Preparedness Exercise

The NRC, along with the Federal Emergency Management Agency (FEMA), DOE, Duke Energy, and the South Carolina Emergency Management Division, provided leadership for the development and conduct of *Southern Exposure 2015*, a fully integrated exercise that demonstrated the nation's ability to effectively respond to a nuclear power plant event that resulted in widespread contamination to the surrounding community. The exercise was conducted in multiple phases over 5 days, beginning on July 21, 2015, and included simulated "time jumps" to 14 days, 6 months, and 18 months postevent. Several workshops and seminars were also held leading up to the exercise. *Southern Exposure 2015* was the first nuclear plant exercise to include long-term recovery elements that focused on housing, agriculture, and economic impact from a catastrophic radiological event. During the exercise, the NRC hosted a number of foreign regulators and observers, including visitors from our neighboring countries.

After the conclusion of the exercise on September 10, 2015, the NRC staff developed the agency's internal after-action report using input collected from evaluators and participants involved in every aspect of the exercise. Key observations from the NRC's internal after-action report include:

- The benefit of continued exercises with the licensee after completion of the inspected portion of the scenario

- The benefit of coordination between the NRC's incident response program and a fully integrated interagency Joint Information Center for public messaging

- The benefit of interagency coordination due to the establishment, by FEMA and the State of South Carolina, of a Unified Coordination Group in response to a nuclear power plant scenario

- The need for improvements in our understanding of, and preparedness for, the NRC's responsibilities in the recovery mission area and fulfillment of the provisions of the Price-Anderson Act that facilitate financial compensation of damages caused by the event

- The need to address limitations of NRC information technology, both in the Headquarters Operations Center and among deployed responders

- The need to more clearly define the roles, responsibilities, and designation process of the senior NRC official deployed to the Unified Coordination Group

- The need to develop a more systematic method for NRC to transfer Federal coordination functions to the U.S. Department of Homeland Security (FEMA), per Federal response and recovery guidance

The NRC staff worked closely with U.S. interagency, State, and private sector partners, primarily FEMA, DOE, the National Nuclear Security Administration, the State of South Carolina, and Duke Energy, to write and issue a combined interagency after-action report. The interagency after-action report and the associated improvement plan were approved and issued on June 21, 2016. Key findings and observations from the interagency after-action report that directly, or indirectly, involve the NRC include:

- The need to define, in Federal guidance, the recommended membership for a Unified Coordination Group established in response to a nuclear power plant incident

- The need to develop a more systematic process to transition responsibility for coordination of Federal response efforts from the NRC to FEMA

- The need for improved awareness and understanding of response and recovery funding mechanisms and methods of financial compensation (i.e., the Price-Anderson Act), which could be available following a radiological event at a nuclear power plant

- The need to improve the organization of the interagency Joint Information Center to facilitate a coordinated message development and approval process

- The fact that disposal of large quantities of radioactive waste will present short and long-term technical and policy challenges

- The value, during all three stages of the exercise, of including the Price-Anderson Act, both as a topic of discussion and as a framework around which to base funding and compensation discussions

- The need to identify a Federal agency as the proponent of remediation in Federal response and recovery guidance

Additional information can be found in Section 16.7 of this report.

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 IRRS and 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 2013 U.S. National Report at the sixth CNS review meeting in April 2014 were very positive.

Items Resulting from Country Group Session

A review of the questions raised by other contracting parties on the 2013 U.S. National Report identified the following areas of interest:

- Fukushima lessons learned
- health physics
- safety culture
- human factors
- digital instrumentation and control
- license renewal and aging
- Reactor Oversight Process
- operating experience
- siting
- materials degradation
- emergency preparedness
- risk-informed regulations and PRA
- licensing and power uprates
- OSARTs
- independence
- decommissioning
- vendor inspection and quality assurance

The NRC's presentation during the 2014 review meeting focused on these topics. INPO, representing the U.S. nuclear industry, also discussed its role in maintaining and improving nuclear safety.

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

- extensive international and national research program in connection with the Fukushima accident
- regulatory openness and transparency
- establishment of a construction experience program and construction resident inspectors offices for new builds
- reintegration of security in the Reactor Oversight Process

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

- Fukushima-related activities (discussed in Sections 1.3.1 and 1.3.3 of this report)
  - completion of most of the Tier 1 recommendations by the end of 2016
  - resource constraints on implementation and verification of modifications according to Tier 2 (external hazards)
  - Tier 3 long-term evaluations
- risk-informed fire protection regulations requiring extensive resources for the transition to National Fire Protection Association 805 (discussed in Section 6.3.8 of this report)
- ensuring continuity during the oversight transition from plant construction to operation (discussed in Section 1.3.2 of this report)
- nuclear industry strategy (discussed in Part 3 of this report)
  - continuous improvement in self-awareness
  - effective use of operating experience
  - maintaining proficiency of the nuclear workforce
  - sharing best practices of supplier and nonnuclear support
  - site resiliency against external events
  - quickly and sustainably recovering lower performing plant
- reporting status of the periodic safety review gap being developed (discussed in Sections 8.5.5 and 14.1.5.7 of this report)
- status of NRC's work on the issuance of possible license renewals to operate beyond 60 years (discussed in Section 14.1.4.3 of this report)

Country Group 1 highlighted the following planned U.S. initiatives:

- Fukushima-related activities (discussed in Sections 1.3.1 and 1.3.3 of this report)
  - work conducted through NEA's Committee on the Safety of Nuclear Installations for identifying gaps that need international work
  - nuclear industry application of lessons learned
- application of risk-informed fire protection regulations for substantial safety improvements (discussed in Section 6.3.8 of this report)
- transition from plant construction to operation (discussed in Section 1.3.2 of this report)
  - coordination of efforts
  - ensure continuity during the oversight transition
- results of Commission's decision on environmental impacts on storing spent nuclear fuel beyond the licensed life for operation (discussed in Section 1.3.3 of this report)

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

Vienna Declaration on Nuclear Safety

Since the Fukushima accident in 2011, the international community has come together to strengthen standards and address lessons learned through a variety of efforts. Some of the most important efforts were led by the CNS Contracting Parties, as evidenced by the work undertaken at the CNS Extraordinary Meeting in 2012, and at the 6th Review Meeting in 2014, to strengthen the CNS guidance documents. In addition, the Contracting Parties convened a CNS Diplomatic Conference, which took place in February 2015. In preparation for the Diplomatic Conference, the Contracting Parties thoroughly considered a proposal to amend Article 18, "Design and Construction," of the Convention. The Contracting Parties agreed not to amend the CNS. At the Diplomatic Conference it was decided to continue moving the Convention forward by recommitting and rededicating ourselves to a vigorous implementation of the CNS. Rather than amending the Convention, the Contracting Parties unanimously adopted the "Vienna Declaration on Nuclear Safety" to reinforce the commitment to meet the Convention's objective to prevent accidents and mitigate their radiological consequences, should they occur. The Vienna Declaration on Nuclear Safety, which is codified in Information Circular (INFCIRC) 872, dated February 18, 2015, states:

- New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.

- Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.

- National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified *inter alia* in the Review Meetings of the CNS.

The Vienna Declaration on Nuclear Safety does not establish new requirements but recommits the Contracting Parties to the implementation of the CNS principles, in particular Articles 6, 14, 17, 18, and 19. The Vienna Declaration on Nuclear Safety is consistent with the CNS reporting guidance, INFCIRC 572, Revision 5, "Guidelines Regarding National Reports under the Convention on Nuclear Safety," dated January 16, 2015.

The United States has consistently addressed the principles documented in the Vienna Declaration on Nuclear Safety since the inception of the CNS. For the purpose of facilitating the peer review to be conducted by the Contracting Parties in preparation for the 2017 CNS Review Meeting, the NRC has included in this report a short summary discussing how the United States addresses the principles of the Vienna Declaration on Nuclear Safety through the implementation of its mature and robust regulatory programs in the aforementioned articles. The Vienna Declaration is discussed in more detail in Sections 6.5, 14.4, 17.6, 18.6, and 19.9 of this report.

1.4.2 Integrated Regulatory Review Service

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

1.4.3 Operational Safety Review Team

The NRC regularly provides technical experts, often at a senior leadership level, to participate in OSART missions around the world. In August 2014, Clinton Power Station, Unit 1, hosted an OSART mission. The OSART team concluded that the managers and the staff of Clinton Power Station are committed to improving the operational safety and reliability of their station. A followup OSART mission was hosted in October 2015, as discussed in greater detail in Section 8.1.5.3 of this report. The next OSART in the United States will take place in 2017 at Sequoyah Nuclear Plant located in Soddy-Daisy, Tennessee.

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. This section also addresses the Vienna Declaration on Nuclear Safety, which was issued in 2015.

The U.S. NRC 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 strategic goals are to ensure the safe and secure use of radioactive materials. Appendix A of this report summarizes the NRC's Strategic Plan, which identifies the agency's strategic goals, objectives, and strategies in more detail.

The agency achieves its safety 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.

The NRC currently uses six performance goals and indicators to track the effectiveness of its nuclear safety regulatory programs to determine whether this goal has been met. Of these six, the following four are related to commercial nuclear power plants:

(1) number of radiation exposures that meet or exceed abnormal occurrence[9] criteria I.A.1, I.A.2, or I.A.3

(2) number of releases of radioactive materials that meet or exceed abnormal occurrence criterion I.B

9 Abnormal occurrence criteria are defined in Appendix A to NUREG-0090, Volume 37, "Report to Congress on Abnormal Occurrences, Fiscal Year 2014," dated May 2015.

(3) number of instances of unintended nuclear chain reactions involving NRC-licensed materials

(4) number of malfunctions, deficiencies, events, or conditions at commercial nuclear power plants (operating or under construction) that meet or exceed abnormal occurrence criteria II.A – II.D

In Fiscal Year (FY) 2015, the NRC met all of its performance indicator targets, and thus, achieved its safety goal strategic objective. The NRC also met its previous performance indicators in FYs 2013 and 2014.

6.2 Nuclear Installations in the United States

Appendix D of this report lists all 100 operating nuclear installations in the United States, as discussed in NUREG-1350, Volume 27, "Information Digest 2015-2016," issued in August 2015, which is available on the agency's Web site. Watts Bar Nuclear Plant, Unit 2, received its operating license on October 22, 2015. Additional information on Watts Bar, Unit 2, can be found in Section 18.1.2 of this report.

Appendix A to 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.

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. These four Westinghouse's Advanced Passive (AP) 1000 reactors are currently under construction. The NRC provides regulatory oversight of their construction using its construction inspection program for units licensed under 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 of this report.

In the Sixth U.S. National Report, the NRC reported that operations had ceased at Crystal River, Unit 3, Kewaunee Power Station, and San Onofre Nuclear Generation Station, Units 2 and 3, in 2013. The NRC also reported that Vermont Yankee Nuclear Power Station would cease power production after its current fuel cycle and move to safe shutdown in 2014. On December 29, 2014, operations ceased at the Vermont Yankee plant.

In 2015, Entergy Corporation announced that it plans to close Pilgrim Nuclear Power Station and the James A. FitzPatrick Nuclear Power Plant. By letter dated November 10, 2015, Entergy Corporation informed the NRC that it will permanently cease power operations at the Pilgrim Nuclear Power Station no later than June 1, 2019. On April 14, 2016, Entergy announced that it would operate Pilgrim for one more fuel cycle and plans to permanently shutdown the reactor on May 31, 2019. By letter dated November 18, 2015, Entergy informed the NRC that it will also permanently cease operations at the James A. FitzPatrick Nuclear Power Plant by the projected end of the current fuel cycle. On March 16, 2016, Entergy formally certified that FitzPatrick would permanently cease operation on January 27, 2017. The company cited low natural gas prices and increased operational costs as key factors in its decision to close the plants.

On June 2, 2016, Exelon Generation also announced its plans to close Clinton Power Station, Unit 1, on June 1, 2017, and Quad Cities Nuclear Power Station, Units 1 and 2, on June 1, 2018. By letters dated June 20, 2016, Exelon notified the NRC that it will permanently cease operations at these three units. The company cited deteriorating economics as the reason for the closures.

On June 16, 2016, the Omaha Public Power District's board voted unanimously to close Fort Calhoun Station, Unit 1, by the end of 2016. The licensee confirmed its decision to permanently cease operations due to financial reasons in a letter to the NRC dated June 24, 2016.

On June 21, 2016, Pacific Gas and Electric Company notified the NRC that it will not pursue license renewal for Diablo Canyon Nuclear Power Plant, Units 1 and 2, and will close the plants in 2024 and 2025, when their existing licenses expire. The decision by the licensee was an agreement in principle not to proceed with license renewal in support of the State of California's policy preference to meet the state's future electricity needs with renewable generation resources, energy efficiency, or storage.

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. After 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 two-step process, specified in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," first issuing a construction permit and then an operating license. Since 1976, the NRC has not received any applications to construct a new reactor under 10 CFR Part 50.

A revised, single-step process was adopted in 1989 and is specified in 10 CFR Part 52, provides direction for issuing a combined license for construction and operation of a new reactor. The NRC has issued seven combined licenses since 2012, authorizing the construction and operation of seven new units at four 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 five 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 five 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, changes of ownership and license transfer, exemptions and relief from NRC regulations, and increases in the reactor power level (i.e., power uprates). Additional information can be found in Articles 14, 17, and 18 of this report.

6.3.2 Reactor Oversight Process

Through its Reactor Oversight Process, the NRC provides continuous oversight of nuclear power plant licensees to verify that they are operating 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 order immediate licensee actions, up to and including a plant shutdown, to address declining or unacceptable safety or security performance at a domestic nuclear power plant.

The Reactor Oversight Process monitors licensee 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 performance across the seven cornerstones using both inspection findings and performance indicators. At least two resident inspectors are stationed at each operating nuclear power plant to monitor plant status, perform routine inspections, and provide immediate response to events. Additional inspectors from the NRC's regional offices and headquarters perform more specialized inspections in areas like fire protection, operator licensing, security, and other aspects of plant design and operation. Each nuclear plant receives risk-informed baseline inspections, which represent the level of NRC inspection required to adequately assess licensee performance. Baseline inspections are used to augment performance indicator data, which is reported quarterly to the NRC to determine licensee performance. The NRC posts plant-specific inspection findings and performance indicator information on the agency's public Web site.

The NRC uses the Reactor Oversight Process Action Matrix to objectively and predictably assess licensee performance and to determine its regulatory response. The Action Matrix classifies licensee performance using five columns, ranging from Column 1, which represents all cornerstone objectives being met, to Column 5, which represents unacceptable performance. Using the Action Matrix, the NRC staff assess licensee performance using inspection finding and performance indicator inputs and directs a graded NRC response to declining performance. Identified inspection findings having more than very low safety or security significance or performance indicators crossing an established threshold may result in supplemental inspections and other possible regulatory actions.

The NRC conducts an annual Agency Action Review Meeting to review the appropriateness of agency actions taken for those power reactor plants with significant performance issues and those that have moved into the "multiple/repetitive degraded cornerstone" or the "unacceptable performance" columns of the Reactor Oversight Process Action Matrix. The Agency Action Review Meeting is an integral part of the evaluative process used by the agency to ensure the operational safety performance of nuclear power plant licensees, and to ensure that trends in industry and licensee performance are recognized and appropriately addressed. After each Agency Action Review Meeting, licensees are informed of any NRC decisions or actions that differ from those previously conveyed (if any agency actions change as a result of the Agency Action Review Meeting). Finally, the Commission is briefed on the Agency Action Review Meeting results at a public meeting.

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. Performance information and additional information about the Reactor Oversight Process can be accessed at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html.

As of August 2, 2016, the Action Matrix assessment of licensee performance at nuclear reactors was as follows:

- Column 1: 89 reactor units in Licensee Response
- Column 2: 8 reactor units in Regulatory Response
- Column 3: No units in Degraded Cornerstone
- Column 4: 3 reactor units in Multiple/Repetitive Degraded Cornerstone

The results of periodic Reactor Oversight Process self-assessments and other independent or focused evaluations have indicated that the program remains effective. An example of the NRC's focused evaluations was the in-depth review of the Reactor Oversight Process conducted in 2014. The results are documented in a report entitled "Reactor Oversight Process Enhancement Project – Baseline Inspection Program," dated April 4, 2014. The goals for that review included enhancing the baseline inspection program to incorporate the inspection areas for the current environment, eliminate redundant inspections, maximize efficient and effective use of resources, and incorporate flexibility where appropriate. Significant outreach to internal and external stakeholders resulted in a comprehensive set of recommendations for each baseline IP.

The NRC temporarily suspended the annual self-assessment process for calendar year 2014 to develop a more effective self-assessment process with more meaningful metrics, and to address Reactor Oversight Process improvement recommendations from multiple independent and focused assessments. In 2015, the NRC staff redesigned the self-assessment process to better assess the effectiveness of a mature program by focusing on the efficacy of recent changes to the program, performing indepth reviews of specific areas of interest, and verifying agency adherence to program governance. The staff will implement the revised process for its calendar year 2016 self-assessment.

The Reactor Oversight Process has developed into a mature oversight process since its inception in 2000; however, the staff recognizes the value of continuous improvement and has actively sought to improve various key program areas through the solicitations of internal and external stakeholder feedback, lessons learned studies, and broader enhancement initiatives.

6.3.3 Industry Trends Program

The NRC staff has implemented the Industry Trends Program since 2001, to confirm safe operation of 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 January 26, 2016, 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 the 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 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 21, "Performance and Accountability Report – Fiscal Year 2015," was issued in November 2015.

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 requiring generic action were identified in FY 2015.

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 2015.

In 2008, the NRC staff implemented the Baseline Risk Index for Initiating Events (BRIIE) as part of the Industry Trends Program. The BRIIE tracks several types of events that could initiate a challenge to a plant's safety systems. 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. None of the initiating events tracked by the BRIIE exceeded its prediction limit in FY 2015.

SECY-16-0044, "Fiscal Year 2015 Results of the Industry Trends Program for Operating Power Reactors," dated April 5, 2016, which is available on the NRC public Web site, provides more details on the Industry Trends Program results for FY 2015.

The Industry Trends Program will be discontinued in 2016, as a result of Project Aim, which is the NRC's effort to develop an integrated prioritization and re-baselining of agency activities. It has been determined that, while the Industry Trends Program provides data that has helped validate broad industry performance trends, no regulatory action has ever resulted from its insights. The NRC has noted that any negative trends in industry performance that the Industry Trends Program could highlight would be self-revealing or identified through other means, such as routine licensee performance assessment, the Reactor Oversight Process self-assessment, end-of-cycle assessment meetings, and the operating experience program.

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). This program provides a comprehensive, risk-informed view of nuclear power plant operating experience and a measure for trending nuclear power plant core damage risk; provides a partial check on dominant core damage scenarios predicted by PRAs; and provides feedback to regulatory activities.

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 NRC also uses the Accident Sequence Precursor Program results to monitor performance against performance indicators in the agency's Congressional Budget Justification and Industry Trends Program, as well as in reports to Congress on events of high safety significance in accordance with abnormal occurrence criteria. Included in the Accident Sequence Precursor Program are the following inputs to programs and reports:

- Number of significant precursor events for the annual Congressional Budget Justification. Accident Sequence Precursor Program results are used as one of several inputs to the performance indicator "Number of malfunctions, deficiencies, events, or conditions at commercial nuclear power plants (operating or under construction) that meet or exceed abnormal occurrence criteria II.A though II.D."

- Description of significant precursor events for the annual abnormal occurrence report to Congress in accordance with Criterion II.C of NUREG-0090, Volume 37, "Report to Congress on Abnormal Occurrences Fiscal Year 2014," dated May 2015.

The staff completed precursor trend analyses as part of the annual Accident Sequence Precursor Program status report provided to the Commission in SECY-15-0124, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated October 5, 2015. The report provided insights such as the following:

- No significant precursors were identified in FY 2014.

- No statistically significant trend was identified for all precursors during the FY 2005 through FY 2014 period.

- In the FY 2012 and FY 2013 annual report, statistically significant increasing trends were identified in the mean occurrence rate of precursors with a conditional core damage probability or Δ core damage probability greater than or equal to 1×10^{-4}. As reported in SECY-14-0107, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models" dated October 6, 2014, six of the seven precursors in this group were caused by multiple electrical failures during a 3-year period. The staff initiated a detailed study in FY 2014 to better understand the contribution of electrical system and associated component failures on risk at U.S. nuclear power plants. Results for this study should be available in FY 2017.

- For the period of FY 2005 through FY 2014, the staff found a statistically significant increasing trend in the mean occurrence rate of precursors resulting from a loss of offsite power initiating event. The staff initiated a detailed study in FY 2014 to better understand the increasing trend. Results for this study should be available in FY 2017.

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 using lessons learned from domestic and international operating experience and other sources 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.

Since the program's launch, the NRC has maintained 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 Web site allows the NRC to quickly disseminate operating experience to the appropriate technical staff. The agency's public Web site at http://www.nrc.gov/reading-rm/doc-collections/event-status/ 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 that the NRC maintain a 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 proposed generic issues include safety evaluations, operational events, and suggestions from NRC staff members, outside organizations, or members of the general public. For emergent issues that demand immediate attention, there are other existing NRC programs to make timely decisions, including issuing immediately effective orders, or if necessary, requiring a plant to shut down.

The Generic Issues Program consists of three stages for processing a generic issue: screening, assessment, and regulatory office implementation. A review panel, consisting of NRC staff with appropriate skill sets, makes the determination if the proposed issue meets the requirements to proceed from one stage to the next. During the screening stage, the proposed issue is evaluated to determine if it satisfies all the following seven screening criteria:

(1) affects public health and safety, security, or the environment
(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) issue is well defined and discrete
(7) may involve review, analysis, or action by the licensee

If the proposed issue meets all the screening criteria, it proceeds forward to the assessment stage. In the assessment stage, the staff evaluates the potential impacts that the proposed issue has on licensees and determines whether the risk is significant enough to warrant additional, or changes to, regulatory requirements or guidance. In the regulatory office implementation stage, the appropriate NRC office develops the necessary regulatory actions to resolve the issue to ensure that adequate safety is maintained at the affected facilities. Depending on the safety significance of the proposed issue, these regulatory actions can include issuing generic communications (e.g., GLs), ordering facilities to backfit changes, and a rulemaking. The Generic Issues Program staff track the status of the generic issue until all required actions are taken and the issue is closed. Additional information regarding the Generic Issues Program can be found in the NRC public Web site at http://www.nrc.gov/about-nrc/regulatory/gen-issues.html, as well as a history of generic issues that is maintained in NUREG-0933, "Resolution of Generic Safety Issues."

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. A congressional mandate, an Executive Order, a petition for rulemaking from outside the NRC, or an internal recommendation from the technical staff may result in the NRC staff's decision to pursue a rulemaking. The NRC recently made changes to enhance Commission involvement in the rulemaking process with the objective of ensuring early Commission engagement before expending significant NRC staff resources. One of the changes requires the NRC staff to prepare a streamlined rulemaking plan before initiating a new rulemaking activity that is not a staff-delegated rulemaking. The Commission reviews this plan and issues its decision (e.g., approval or denial) in the form of a Staff Requirements Memorandum. Another change is that the NRC staff will request approval from the Commission to discontinue or delay rulemaking activities. The staff may request that a rulemaking activity be discontinued or delayed at any stage in the rulemaking process. As a result of recent Project Aim rebaselining evaluations, the NRC has discontinued or delayed a number of low-priority rulemaking activities in various stages of development.

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 for consideration. Generally, all rules are issued for public comment. Those rules exempted 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. Section 1.3.3 of this report 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 two main foci in fire protection regulation: (1) implementation of the new risk-informed, performance-based fire protection licensing basis (10 CFR 50.48(c)); and (2) resolution of the fire-induced multiple spurious operation and circuit analysis issue.

To support the implementation of 10 CFR 50.48(c), the NRC issued RG 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," dated December 2009, 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 July 2016, 28 plants, representing 44 units submitted license applications to transition to 10 CFR 50.48(c). Nineteen license amendments have been issued, and nine are still under review. One additional license application representing two units is expected.

Challenges associated with the completion of the 10 CFR 50.48(c) reviews include: licensee-initiated rework, lengthy response times to requests for additional information, licensee-initiated technical changes to their license application within the final weeks of the review, increased technical complexity, and licensee use of new or refined methods that were not included in previously issued regulatory guidance. The staff overcame these challenges but they impacted the schedule for completing the safety evaluation reports.

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 in October 2009, provides regulatory guidance for licensees on fire protection issues, including the treatment of fire-induced circuit failures in response to fire damage. The NRC staff is working with industry stakeholders to enhance guidance regarding fire-induced multiple spurious operations through the planned development of Volume 3 to NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE).”

Plants that are not transitioning, or have not completed their transition, to 10 CFR 50.48(c) are inspected under IP 71111.05T, “Fire Protection (Triennial),” dated January 31, 2013. Plants that have completed their transition to 10 CFR 50.48(c) are inspected under IP 71111.05XT, “Fire Protection – NFPA 805 (Triennial),” dated April 19, 2011. Findings identified for licensees under both regulatory frameworks are evaluated using Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” dated September 20, 2013. 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 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 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, “Standards for Protection against Radiation”). 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 in 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 in 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.

The NRC has determined that spent fuel can safely remain stored in the SFPs or in dry cask storage facilities until a geologic repository is built and operating (see Section 1.3.3 of this report for further details on Continued Storage). 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.

The current NRC reactor decommissioning requirements have been safely implemented. Since the early 1980s, 10 power reactors have been decommissioned and their licenses terminated. However, reactors undergoing decommissioning required several exemptions from the NRC's regulations to reflect their decommissioning status. As discussed in Section 1.3.2 of this report, the NRC has initiated a new decommissioning rulemaking process that may seek to provide an efficient, open, and clear decommissioning process by reducing the need for exemptions from existing regulations, addressing the appropriateness of maintaining these existing options for decommissioning and the 60-year timeframe for decommissioning, evaluating the advisability of requiring a licensee's Post-Shutdown Decommissioning Activity Report to be approved by the NRC, and determining the appropriate role of state and local governments and non-governmental stakeholders in the decommissioning process.

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 preapplication reviews for advanced nonlight-water reactor designs. In the preapplication 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 the Regulation of Advanced Reactors," (73 FR 60612, dated October 14, 2008), 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 believes that nuclear regulation should be conducted as openly as possible. Ensuring appropriate openness explicitly recognizes 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, Facebook, and the NRC blog) to disseminate timely, accurate information on issues of interest to the public or events at nuclear facilities. The NRC seeks public involvement early in the regulatory process to promptly address any safety concerns. 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 the 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 thousands of documents related to rulemaking actions that the NRC has conducted from May 1996 to the present. 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 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 that the public, industry, and NRC employees can use to raise safety concerns (as discussed in Section 10.4.2 of this report), 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 petition 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. If the petition for rulemaking meets the NRC's requirements for docketing, then the NRC usually publishes a notice of docketing of the petition in the *Federal Register*.[10] When the NRC seeks additional information or opinions to help resolve the petition for rulemaking, that notice of docketing offers a public comment period. The NRC evaluates the petition and any comments received and may either determine to consider the petition in a current or future rulemaking ("enter the issues into the rulemaking process") or deny the petition (in its entirety or in part). 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.

If the NRC decides to enter the issues into the rulemaking process, the NRC then addresses the issues in the same manner and using the same criteria (e.g., risk significance, costs and benefits as evaluated in a regulatory analysis, backfitting and issue finality) as any other issue that is the subject of rulemaking. If the NRC believes that rulemaking action is justified, then the NRC publishes a proposed rule addressing the issues raised in the petition. This action is followed by a public comment period and publication of a final rule. If, as a result of the rulemaking process, the NRC decides not to take action addressing some or all of the issues in the petition for rulemaking, then the NRC publishes a notice setting forth the reasons for deciding not to take action on those issues originally raised in the petition; this constitutes the "final denial" of the petition for rulemaking with respect to those issues.

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 (76 FR 34773, dated June 14, 2011), safety conscious work environment guidance documents, and other related regulatory tools such as safety culture case studies.

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10 In some circumstances, the NRC may determine that the issues raised in the petition for rulemaking should be immediately considered in a rulemaking. In such cases, the NRC places the petition's issues in the rulemaking process without opening a petition for rulemaking docket, and publishes a notice of that action in the *Federal Register*.

Additionally, workers and members of the public may bring their concerns about safety or regulatory issues directly to the NRC. The agency maintains a tollfree 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 500 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 evaluates each issue to determine whether it can verify the issue and, if so, the effect of the issue on public 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 20 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 shut down, if necessary. The NRC took prompt action following the Fukushima accident through the issuance of orders, implementation of focused inspections, development of INs to the industry, and issuance of bulletins to confirm that there were no imminent 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, inspections, use of operating experience, rulemaking, and conducting additional research.

6.5 Vienna Declaration on Nuclear Safety

The mission of the NRC is to protect public health and safety, and the environment. The NRC's primary goals are ensuring the safe and secure use of radioactive materials. The agency achieves this goal by ensuring that licensee performance is at or above acceptable safety levels. As discussed in Section 6.1, 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 to:

- Prevent and mitigate accidents and ensure radiation safety
- Ensure protection of nuclear facilities and radioactive materials

Nuclear power plants must meet the NRC's safety, security, technical and financial qualification requirements codified in 10 CFR Chapter I. These regulations serve to prevent accidents and mitigate adverse consequences in a manner that effectively minimizes the potential for (and therefore addresses the risk of adverse consequences associated with) unintended releases of radioactive materials. Because NRC requirements protect public health and safety through prevention of accidents and by mitigating releases in the event of an accident, the risk of offsite contamination is rendered acceptably low as an indirect benefit, rather than as a direct performance goal.

The NRC uses deterministic, risk-informed, performance-based, and defense-in-depth requirements to achieve this goal. The NRC's defense-in-depth philosophy includes: (1) the need to prevent accidents from occurring and mitigating accidents if they occur, (2) the concept of multiple barriers against radioactive releases, and (3) the application of the principles of independence, redundancy and diversity, which is implemented through requirements such as the "single failure" assumption. Thus, the NRC's current regulatory approach provides reasonable assurance that an accident resulting in long-term offsite contamination is unlikely. Section 18.1 of this report discusses the NRC's defense-in-depth philosophy in more detail.

As described in Section 14.1.5.1 of this report, the NRC carries out many regulatory activities that, when considered together, constitute a process providing ongoing reasonable assurance that the licensing bases of nuclear power plants provide adequate protection of public health and 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 other regulatory actions to resolve identified issues.

In light of the Fukushima accident, the NRC has taken many actions to strengthen the protection of U.S. nuclear plants against events that could exceed a plant's design basis. For example, the NRC issued regulatory requirements, in the form of three orders, based on the lessons learned from Fukushima. The three orders required safety enhancements of operating reactors, construction permit holders, and combined license holders. These orders required nuclear power plants to implement safety enhancements related to: (1) mitigation strategies to respond to external events resulting in the loss of all AC power at plants, (2) ensuring reliable severe accident capable hardened containment vents for Mark I and II boiling-water reactors designs, and (3) enhancing SFP instrumentation. Operating plants were required to begin implementation of the safety enhancements promptly and complete implementation within two refueling outages or by December 31, 2016, whichever came first. In the case of the containment vents, the NRC revised its original order for hardened vents for BWRs with Mark I and II containments to include additional requirements for those vents to have capabilities to be operated under severe accident conditions. This revision resulted in a change to the required date for full implementation to be achieved. The NRC has taken other actions to address the adequacy of nuclear power plant design with respect to natural hazards (e.g., seismicity, and flooding). The NRC is continuing its post-Fukushima activities through the development of a new regulation for mitigating beyond-design basis events. The NRC's response to Fukushima reflects the NRC's regulatory approach of promptly addressing potentially significant safety issues at the time they are discovered, and taking appropriate action in a timely fashion, rather than awaiting a periodic review. Section 1.3.1 of this report provides a complete description of NRC post-Fukushima activities.

Additionally, as described in Section 16.5 of this report, the NRC's Reactor Oversight Process specifically addresses emergency preparedness, and the NRC's enforcement program, which is described in Section 9.3 of this report, ensures that safety improvements are implemented promptly. Furthermore, at each license renewal of a nuclear power plant, the NRC performs a safety review of systems, structures and components that the NRC has concluded require a formal review. A description of the NRC's license renewal program is discussed in Section 14.1.4 of this report. Furthermore, a detailed description of the NRC's regulatory approach in ensuring that adequate protection continues to be provided by nuclear power plants on a continuing basis throughout operation is set forth in NUREG-1412, "Foundation of the Adequacy of the Licensing Basis," issued in December 1991.

Finally, the NRC actively participates in the development of the IAEA safety standards. The NRC is represented at the IAEA Commission on Safety Standards and all IAEA Safety Standards Committees by senior executive managers. The NRC also ensures that NRC regulations are consistent with IAEA safety standards. In addition, the NRC reviews IAEA safety standards when revising guidance in regulatory guides, to gain insights. Additional information about how the IAEA safety standards are used at the NRC can be found in Section 8.1.5.1 of this report.

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.

The United States did not change the legislative framework governing the U.S. nuclear industry as a result of the Fukushima accident. The U.S. NRC has taken the necessary regulatory actions in response to the accident, as described in Sections 1.3.1 and 1.3.3 of this report, under the existing framework.

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 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. 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 U.S.-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 Amendment to the Convention on the Physical Protection of Nuclear Material, ratified in 2015, strengthens obligations for the physical protection of nuclear material in domestic use, storage, and transport, and for the protection of nuclear material and nuclear facilities from sabotage.

- 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 U.S. 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 Federal Emergency Management Agency (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, 10 CFR, Chapter I, governs, among other things, the licensing of nuclear installations. The NRC established these regulations through informal, "notice-and-comment" rulemaking procedures under the Administrative Procedure Act. In short, these rulemaking procedures typically include: (1) publishing a proposed rule for public comment; (2) after considering comments, providing public notice of the issuance of the final rule and an effective date for the final rule; and (3) including a statement of the rule's basis and purpose. Once these final rules are in place, they are binding on the applicable regulated entities (including operators of nuclear installations) and can be substantively 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 is responsible for licensing of all commercial and industrial nuclear production and utilization facilities or installations, including nuclear power reactors, in the United States. As discussed in Section 8.1.2.1 of this report, 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 production and utilization facility without a valid license issued by the NRC. Section 103, which applies to facilities for industrial or commercial purposes, 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 an opportunity for hearing in proceedings for the granting, suspending, revoking, or amending of licenses (including renewed operating licenses and construction permits for 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 comprised of one lawyer and two technical members, but hearings 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, "Domestic Licensing of Production and Utilization Facilities," 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, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” which provides for combined licenses that resolve all safety issues before construction, and early site permits that can resolve most siting issues separate from a license application. The basic concept underlying 10 CFR Part 52 is to provide for early resolution of licensing issues by approving nuclear reactor designs through generic rulemaking (design certification). 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 design certification rulemaking.

Under the combined license process in 10 CFR Part 52, the NRC determines and approves, 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.

The initial license for a nuclear power plant may be renewed 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.”

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 verify that the licensee is safely operating the facility. The agency integrates inspection results into its overall evaluation of licensee performance, as discussed in Article 6 of this report. As described in Section 7.2.4 of this report, the NRC may take enforcement action to address safety and security concerns and/or violations of NRC requirements.

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 during the annual Agency Action Review Meeting and report these results in a public Commission meeting. This meeting provides another opportunity to discuss significant events, licensee performance issues, trends, and actions to mitigate recurrences. This is further discussed in Section 6.3.2 of this report.

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 Attorney General 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 imposition of 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 U.S. government to impose criminal penalties on persons who interfere with nuclear inspectors. Section 236 allows the imposition of 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 of this report.

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 U.S. 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., the U.S. DOE). It discusses financial and human resources aspects, the regulatory body's international responsibilities, its ethics rules, and its policy for maintaining openness and transparency.

The United States did not change the legislative framework governing the U.S. nuclear industry as a result of the Fukushima accident. The NRC has taken the necessary regulatory actions in response to the accident, including the creation of the Japan Lessons Learned Division, as described in Sections 1.3.1 and 1.3.3 of this report.

8.1 The Regulatory Body

This section explains the NRC's mandate, authority and responsibilities, structure and position in the Government, its financial and human resources, as well as its 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.

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, 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, judicial review of Commission decisions is limited. Courts generally defer to the Commission's legal and factual determinations, particularly where they fall within the agency's responsibility for and expertise in nuclear safety.

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. Reorganization Plan No. 1 of 1980 strengthened the executive and administrative roles of the NRC Chairman, particularly in emergencies, while providing that all policy formulation, policy-related rulemaking, and orders and adjudications would remain vested with the full Commission. The Commission as a whole formulates policies and regulations governing safety and security, 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 of decisions made by the Atomic Safety and Licensing Boards. The office provides the Commission with an analysis of adjudicatory matters that may merit a Commission decision, and drafts adjudicatory decisions under the Commission's guidance. The office also supports the Commission when it conducts mandatory hearings associated with certain applications (for example, combined license applications).

- Office of Congressional Affairs. The Office of Congressional Affairs reports solely to the Chairman and is the primary point of contact for all communications between the NRC and Congress. This office provides advice and assistance to the Chairman, the Executive Director for Operations, 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. The NRC Protocol Office and the Federal and External Affairs program also reside in the Office of Congressional Affairs.

- 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 recommendations to the Chairman, the Commission, and the NRC staff on international policy and outreach activities. It plans, develops, and implements programs to carry out statutorily mandated activities in the international arena, including implementation of relevant U.S. treaty obligations and export and import licensing responsibilities. It also 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 reports solely to the Chairman and administers 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. Since the issuance of the previous U.S. National Report, the following major office reorganizations have taken place:

- consolidation of the Office of Federal and State Materials and Environmental Management Programs into the currently existing Office of Nuclear Material Safety and Safeguards

- consolidation of the Computer Security Office and the Office of Information Services into the newly created Office of the Chief Information Officer

The NRC offices are briefly described below.

- Office of Administration. The Office of Administration provides centralized services in the areas of contracts, facilities management, personnel and facilities 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 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 the Chief Information Officer. The Office of the Chief Information Officer plans, directs, and oversees the resources to ensure the delivery of information technology and information management services that are critical to support the mission, goals, and priorities of the agency. In addition, it plans, directs, and oversees the implementation of a comprehensive NRC information technology security program, and also coordinates and oversees the development and update of agencywide information resources management policy.

- Office of Investigations. The Office of Investigations develops policy, procedures, and quality control standards for investigations of licensees, applicants, and their contractors or vendors, including conducting investigations of all allegations of wrongdoing by other than NRC employees and contractors. The Office of Investigations may self-initiate investigations. It makes referrals of substantiated criminal cases to the U.S. Department of Justice for prosecution consideration and coordinates with other agencies and organizations to ensure timely exchange of information of mutual interest. In addition, the Office of Investigations maintains current awareness of inquiries and formal investigations and keeps the Commission informed of matters under investigation as they affect public health and safety, the common defense and security, and the environment.

- 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants," including 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; the transportation of radioactive materials regulated under the Atomic Energy Act; the safe and secure use of radioactive materials in medical, industrial, and academic applications for beneficial civilian purposes; the safe management and disposal of low-level waste; implementing safe materials, power and nonpower reactor decommissioning programs, and the cleanup of contaminated sites; and uranium recovery activities.

- 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 and emergency preparedness at nuclear facilities. The office is the agency's security, emergency preparedness, and incident response interface with other Federal agencies.

- 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.

- 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.

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.

- Advisory Committee on Reactor Safeguards. The Advisory Committee on Reactor Safeguards has statutory responsibilities as described in the Atomic Energy Act of 1954, as amended. The Committee 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 staff 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 analyzes and comments on the impact of existing and proposed legislation and regulations on the economy and efficiency of NRC programs and operations. The Inspector General 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 that have the most frequent contact and interaction with the NRC are the White House, Office of Management and Budget (OMB), U.S. Department of State, DOE, U.S. Environmental Protection Agency (EPA), U.S. Department of Homeland Security (DHS), the 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:

- <u>The White House</u>. As noted in Section 8.1.2.2 of this report, 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.

  Under the aegis of the White House, the National Security Council is tasked with coordinating Executive Branch policies and activities. Through the Interagency Policy

Coordinating committee structure, the NRC and other agencies are able to ensure that program activities are aligned with U.S. foreign policy objectives.

- 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 supports the U.S. Department of State during negotiation of international agreements in the nuclear field and coordinates a number of interactions with IAEA and other international organizations of the United Nations, as well as the 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 EPA intersect or overlap in areas in which EPA issues generally applicable environmental standards for activities that are subject to NRC licensing actions. Examples include general standards for high-level waste repositories, uranium milling facilities, decommissioning standards, and standards for public and worker protection. EPA 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 OMB for approval.

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

The Atomic Energy Act confers on the NRC preemptive authority over health and safety regulation of nuclear energy and radioactive 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 its regulatory 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 the operations of their regulatory programs. However, the NRC does provide technical training to Agreement State staff in order to ensure a more consistent and robust National Materials Program. 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 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 and security standards. Some activities are carried out within the programs of the IAEA, the NEA, or other international organizations. The NRC conducts other activities directly with counterpart agencies in other countries under technical information exchange cooperation arrangements.

International Treaties. Treaties that legally bind 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 members regularly participate in implementation activities related to these conventions and have held a variety of leadership positions at meetings of Contracting Parties. 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, including urging ratification by new states.

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 in 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 National Security Council, 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. In addition to staff participation in more than 200 IAEA and NEA meetings each year, the United States has participated 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 INPO in this process. In 2017, the United States will host an OSART mission at the Sequoyah Nuclear Plant. The United States intends to continue to plan for hosting an OSART mission every 3 years.

Since 1999, the NRC has supported the IRRS program. In recent years, given the increased emphasis on peer reviews following the Fukushima Dai-ichi accident, the NRC has participated in about 10 IRRS 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 was completed in February 2014. Additional information about the 2014 IRRS followup mission's findings can be found in Section 8.1.5.2 of this report.

The NRC actively participates in the IAEA Commission on Safety Standards and all of the IAEA Safety Standards Committees, including the Nuclear Security Guidance Committee. In November 2015, the U.S. Government was appointed chair of the new Emergency Preparedness and Response Standards Committee. 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 serves 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. Furthermore, the NRC is 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 AP1000, Korea Electric Power Corporation and Korea Hydro and Nuclear Power Co., Ltd.'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.

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. The NRC's activities associated to the IAEA Action Plan on Nuclear Safety is discussed in Section 1.3.1 of this report.

In addition, the U.S. Government 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. The U.S. Government also participated in the drafting of the IAEA's report, "The Fukushima Daiichi

Accident," published in 2015, which summarizes both the accident events and actions taken globally to enhance safety. Details of the NRC's evaluation of Fukushima lessons learned can be found in Section 1.3.1 of this report.

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 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 consultation with the IAEA, the NRC has coordinated its International Regulatory Development Partnership program. The International Regulatory Development Partnership provides training in nuclear orientation (codes and standards, fundaments of reactor regulation and safety, PRA quality, and quality assurance), agency infrastructure development (nuclear executive workshop and safety culture), regulatory program development (construction permit application review and site application review), and regulatory process (construction and vendor inspection practices, licensing review methodology and power uprates).

Members of the Commission travel internationally to engage in bilateral exchanges of technical information and attend conferences to deliver keynote remarks, participate in panel discussions, and otherwise share insights on a variety of topics with diverse technical and political audiences. The NRC's annual Regulatory Information Conference also provides a forum for the Commission and NRC staff to hold technical exchanges and high-level bilateral meetings, with more than 30 countries represented each year, many at senior levels.

Bilateral Relations. The NRC has arrangements to exchange technical information with nuclear safety agencies in 45 countries, Taiwan, and the European Atomic Energy Community. In addition, the NRC works with many other countries either bilaterally or regionally 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 methodological regulatory information. The NRC provides advice, training, and other assistance to countries that seek U.S. help to improve their regulatory programs.

Combined with the NRC's export/import licensing outreach activities and work through the various conventions and treaties, 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) provide the U.S. Government with insights into whether national export/import licensing regimes will ensure the safe and secure use of nuclear facilities and materials, (2) alert the U.S. Government and industry to potential safety problems, (3) help find possible accident precursors, and (4) 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 suite of international activities – treaty implementation, export/import licensing and 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 technical information exchange arrangements, including results from the NRC's lessons learned activities which are further addressed in Section 1.3.1 of this report. 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.

International Assistance Programs. Since the early 1990s, the NRC has continued to expand its program of assistance to countries developing nuclear power programs. The NRC began offering assistance to nuclear regulatory programs in several former Soviet states, focusing on countries in which Soviet-designed reactors were operated. After the September 11, 2001, terrorist attacks, the NRC expanded its assistance efforts specifically to 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, consistent with international legal commitments, IAEA standards guidance, and coordinated with other states. 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.

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 and regulatory guidance.

NRC senior managers represent the agency at the IAEA Commission on Safety Standards and all IAEA Safety Standards Committees. 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 and regulatory guidance 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.

Many of the differences in how the safety standards are applied to NRC regulations stem from the fact that NRC regulatory guidance predates most IAEA safety standards. Furthermore, 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 RGs are updated. The NRC has revised its policy guidance and now directs staff to consider

IAEA standards as a point of reference when drafting or revising RGs, and to consider direct endorsement of the IAEA standards where appropriate. Because of this guidance, the NRC has published 13 new or revised RGs that harmonize with or reference IAEA safety standards in the past 2 years.

8.1.5.2 Integrated Regulatory Review Service Mission

The NRC hosted an IRRS mission in October 2010 focused on the U.S. operating power reactor program. The 2010 mission identified 2 recommendations, 20 suggestions, and 25 good practices. Subsequently, the NRC developed an action plan to address the team's findings and hosted a followup mission in 2014. The followup mission also reviewed the NRC's response to the Fukushima accident.

One of the mission's suggestions relates to the conduct of periodic safety reviews. In preparation for the 2010 IRRS mission, the NRC correlated its regulatory programs to the 14 safety factors identified in IAEA Specific Safety Guide SSG-25, "Periodic Safety Review of Nuclear Power Plants Safety," published in 2013. The NRC's objective was to demonstrate clearly that the agency's programs robustly meet the intent of the periodic safety review. The IRRS team concluded that "… the NRC has in place a number of programmes (the analysis of the operating experience, the Reactor Oversight Process, the generic upgrades and regulatory changes, the use of risk informed regulation and the license renewal rule) that are intended to ensure that the goals of the periodic safety review are met and that provide adequate protection to the health and safety of the public, as required by the Atomic Energy Act." Furthermore, the report states: "Although the NRC utilizes an alternate approach to meet the [periodic safety review] PSR safety factors, NRC should incorporate lessons learned from Periodic Safety Reviews performed in other countries as an input to the NRC's assessment processes." To address this, the NRC began a limited scope pilot study that reviewed several periodic safety review reports from other countries to identify areas that could potentially inform the NRC's regulatory processes.

During the 2014 IRRS followup mission, the team closed this item based on progress made by the NRC in the evaluation and incorporation of periodic safety review lessons learned. Upon completion of the pilot study the NRC issued a report entitled, "Findings from the Staff's Evaluation of Periodic Safety Reviews from Other Countries," dated April 24, 2015. The report concluded that it is reasonable to expect that the U.S. regulatory approach would be sufficient for detecting and correcting the plant-specific issues addressed by other countries' periodic safety reviews if they were to occur in U.S. plants. Hence, changes to the existing regulatory processes were deemed unnecessary. Additional information on the periodic safety reviews can be found in Section 14.1.5 of this report.

Relative to the NRC's response to the Fukushima accident, the followup mission concluded in its report that the NRC has "acted promptly and effectively in the interests of the public health and safety in both the U.S. and Japan." The team said the NRC's NTTF report was "a source of inspiration for many regulatory bodies worldwide." The team also reviewed how the NRC inspected U.S. reactors on Fukushima related issues, and called that work "exemplary."

The 2014 IRRS followup mission also closed 1 of the 2 recommendations and 18 additional suggestions for a total of 19 closed suggestions. One new suggestion was opened concerning transition of operating reactor plants to decommissioning. The NRC has continued to make strides on the one recommendation and two suggestions that were outstanding. On April 13, 2016, the United States sent a letter to IAEA that served as the final update regarding the 2010 and 2014 IRRS missions. In summary, the letter, which is available in the NRC's public Web site (ADAMS Accession No. ML16106A037), provided the status of the three outstanding items as follows:

(1) Recommendation 2 —Develop a Methodology and Implement a Holistic Management System Review. The NRC is continuing the development of the remaining process maps for the Operating Reactor Program. After the completion of the process map, the NRC will establish and implement a process for periodic, holistic reviews of the effectiveness of the management system.

(2) Suggestion 7 —Direct Implementation of the term "As Low as Reasonably Achievable (ALARA)." The Commission directed the NRC staff to examine regulations that contain dose criteria and the NRC staff issued an advance notice of proposed rulemaking on July 25, 2014, which discusses ALARA planning. The staff is continuing to evaluate the effectiveness of ALARA planning and will evaluate the public comments on the program to update the guidance documents and determine if other changes to the regulatory framework are appropriate. This initiative is further discussed in Section 15.1 of this report.

(3) Suggestion Followup 1 —Develop a Consolidated Rulemaking and Corresponding Guidance to Facilitate the Orderly Transition from Plant Operation to Plant Decommissioning. The Commission directed the staff to proceed with a rulemaking on reactor decommissioning and set an objective of early 2019 for completion of this rulemaking. The staff has entered into the regulatory basis development stage for the proposed rulemaking entitled "Regulatory Improvements for Power Reactors Transitioning to Decommissioning." This initiative is further discussed in Section 1.3.2 of this report.

The report, IAEA-NS-2014/01, "Integrated Regulatory Review Service (IRRS) Follow-up Mission to the United States of America," published in 2014, is available in the NRC's public Web site (ADAMS Accession No. ML14265A068).

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 licensees that host OSART missions have some reduced NRC inspections under the Reactor Oversight Process.

In August 2014, Clinton Power Station, Unit 1, in Illinois hosted an OSART mission. As written in the report; "The OSART team concluded that the managers and the staff of Clinton Power Station are committed to improving the operational safety and reliability of their station." Several proposals for improvement were made including, "Improve the backlog management tool and methodology so as to ensure timely completion of maintenance work orders even for lower priority work." The IAEA "Draft Report of the Operational Safety Review Team (OSART) Mission to Clinton Power Station," published in December 2014, and is available in the NRC's public Web site (ADAMS Accession No. ML15062A115).

Subsequently, Clinton hosted a followup OSART mission in October 2015. The Clinton OSART report for the followup mission is publicly available in the NRC's Web site (ADAMS Accession No. ML16105A282). The NRC reviewed both reports and did not identify issues pertaining to either the plant or to the NRC's requirements that necessitate NRC program changes.

The Sequoyah Nuclear Plant in Tennessee will host an OSART mission in 2017.

8.1.6 Financial and Human Resources

8.1.6.1 Financial Resources

As of September 30, 2015, 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 FY 2015 enacted budget was $1.0153 billion, including the budget for the Office of the Inspector General. This is a decrease of $40.6 million when compared to the FY 2014 enacted budget of $1.0559 billion.

The NRC FY 2015 budget was financed with $895.5 million from user fees and $119.8 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.

The NRC regularly solicits feedback in an effort to gain independent and diverse perspectives on ways to improve NRC's work environment. In view of that, the agency often explores various channels that seek to provide meaningful insights about employees and their work experience. One such mechanism is our workforce surveys. The NRC participates in two workforce surveys measuring employee perceptions of the work environment: the U.S. Office of Personnel Management Federal Employee Viewpoint Survey and the NRC Safety Culture and Climate Survey. The Federal Employee Viewpoint Survey is mandated by Office of Personnel Management's regulations and is conducted annually. The Safety Culture and Climate Survey is administered by NRC's Office of the Inspector General approximately every three years. These surveys provide unique, but also overlapping, insights on the NRC workplace that together build a comprehensive picture of employees' experiences with their job, supervisors, and work units.

The Federal Employee Viewpoint Survey and the NRC Safety Culture and Climate Survey were administered for the first time in the same year beginning in 2012. The most recent surveys were administered in 2015. Both surveys have consistently revealed that the NRC is a top performing organization within the public sector and ranks competitively against private sector benchmarks.

Although the survey questions on the Federal Employee Viewpoint Survey differ from that of the Safety Culture and Climate Survey, the overall objectives and reason for doing these surveys remain the same: to create an engaging work environment where employees feel that they have opportunities to excel.

Top positive results for the 2015 Federal Employee Viewpoint Survey include willingness to put in the extra effort to get a job done, supervisors talking with staff about performance, and continually looking for ways to do a better job. Opportunities for improvement remain in the areas of paying raises depending on how well employees perform their jobs, steps taken to deal with poor performers, and opportunities to get a better job in your organization.

Strengths from the 2015 Safety Culture and Climate Survey were noted in the areas of mission and objectives, training, and supervision. Areas for improvement included the differing views processes, empowerment and respect, and senior management.

The agency will focus action planning on areas identified in both surveys, along with reinforcing the existence of a positive 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 and lower than projected numbers of new reactors. 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. 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, which allows the staff members to work remotely and reduce their commute times. The agency strives to create a positive organizational culture with an emphasis on a strong safety culture 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 implements a robust onboarding program, including an online employee orientation toolkit. This orientation toolkit allows new hires to access 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 inspectors, technical reviewers, and project managers.

As an example, an increase in development of risk-informed licensing and regulatory applications has created a demand for PRA analysts at both the licensees and the regulator. The NRC 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 now in its fourth year of application at the NRC.

The NRC continues to implement blended learning strategies in the training program. Blended learning is defined as using a combination of educational techniques to optimize knowledge transfer and delivery using both formal and informal approaches. Examples of various educational techniques used at the NRC include classroom instruction, videos, Web sites, virtual classrooms, discussion boards, modeling and simulation, webinars, communities of practice, and hands-on application of knowledge and practice of skills with the support and guidance of a mentor. Benefits of incorporating blended learning include the ability for learners to gain or improve knowledge at any time and incorporate skills practice on the job, which directly decreases the time to competency for employees while saving the agency money by reducing travel costs associated with training attendance, and improving staff productivity by reducing their time away from work.

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 remains a top priority and is an integrated part of the agency's Strategic Plan to ensure we capture and preserve knowledge to assist with employee development and performance. There are four contributing activities:

(1) Provide innovative agency support structures for knowledge management.

(2) Create communities of practice that enable the sharing of relevant knowledge and critical skills among employees who perform the same job function.

(3) Capture operating experience, new information on safety and security issues, and knowledge gained from inspection, research, and licensing activities in regulatory guidance.

(4) Capture relevant critical knowledge from employees departing the agency, recapture knowledge from former employees where possible, communicate leadership expectations for knowledge sharing, formalize knowledge management values and principles, and incorporate knowledge management practices within agency work processes.

A key element to the Knowledge Management Program success is the system of governance under the agency knowledge management Steering Committee and knowledge management staff leads with program management provided by the Office of the Chief Human Capital Officer. These entities oversee and implement activities across the agency ensuring current and future knowledge management needs of the agency are met. To accomplish this, the NRC uses a broad and continuously evolving range of knowledge management tools and methods.

A few examples of current knowledge management and knowledge transfer activities include the following:

- KNOWvember. The month of November is marketed agencywide as *KNOWvember* to raise awareness and provide an opportunity to remind employees of the importance of knowledge management. Every year since its inception in 2010, the program focuses on a different theme to support knowledge management around the agency. In 2015, the agency held a number of sessions with NRC experts highlighting critical skills, topics, and historical events, such as a session on the issuance of NUREG-75/014, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400)," dated October 1975, and the origins of PRA in the nuclear industry. In addition, the agency pursued discussion around the influential topic of multigenerations in the workforce, where a panel session of nine NRC employees from each of the five generations presented their perspectives on "Leading across Generations." All sessions were recorded and preserved for knowledge management. In 2014, the NRC held a 2-day knowledge management best practice showcase that was dedicated to sharing best practices around the agency. The showcase exhibited 20 practices from eight offices.

- Ask SME and Learn. In November 2013, the program launched an internal series called, "Ask SME [subject matter experts] and Learn," that was initiated to capture and share critical knowledge and experiences of subject matter experts. The sessions provide an opportunity for staff across the agency to learn directly from the agency experts on a particular topic in an open forum. Previous sessions that have been held featured senior executives sharing their knowledge and experiences prior to retirement, other topics included cyber security, Price Anderson Act, and low level waste regulation. The sessions are recorded and preserved for knowledge management.

- Knowledge Management Guidance. In October 2015, the program issued recommendations and guidance for sharing knowledge and experience gained by NRC personnel returning from foreign assignments.

- NUREG/KM Series. In 2012, the agency launched a NUREG/Knowledge Management (NUREG/KM) series to preserve knowledge of historical events that shaped the regulatory process. The series focuses on collecting and interpreting historical information on identified topics for the benefit of future generations of NRC professionals as well as the public. Currently, the series features eight publications. The first of the

NUREG/KM series is an account of the Three Mile Island accident. The eighth and most recent series is NUREG/KM-0008, "Reflections on Fukushima: NRC Senior Leadership Visit to Japan, 2014," dated December 2014.

- Brown Bag Sessions (Chat n' Learn). 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. More recently, departing senior executives, such as former Executives Directors for Operations and Office Directors, have used this as a forum to reach different audiences agencywide to share their knowledge and insight on a variety of technical topics, issues, and events they have experienced throughout their careers.

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 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 one of the five "Principles of Good Regulation" 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 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 three specific strategies to achieve openness, including the need to provide meaningful opportunities for the public and other stakeholders to participate in NRC activities 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 comprises 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 for them to be withheld. The agency implemented policies, performance measures, and management controls to ensure this access. The NRC's documents database, known as 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. In 2015, approximately 69,000 public users accessed ADAMS more than 247,000 times and requested documents more than 45.2 million times.

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 and engage the public. The plan lays out specific improvements the NRC is making to enhance stakeholder engagement.

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 where 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 uses its public Web site to share information with stakeholders and the public. In 2015, the NRC's Web site had more than 2.5 million individual visitors. The Web site was visited more than 6.4 million times, and visitors viewed more than 71.4 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 locate information easily 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 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 major 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 receive automatically a number of different types of documents, including generic communications, new rulemaking dockets, speeches, and reports issued by the NRC's Inspector General. The public also can subscribe to receive correspondence related to specific facilities. 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 agency has also made use of webinars as a way of leveraging technology to communicate with the public.

Social Media. Over the past several years, the NRC has embraced social media as an important tool for reaching a public audience beyond those with access to traditional communications media. 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 members at all levels help ensure the agency is meeting the communication needs of all our offices, both at headquarters and in the regions.

The NRC's blog, or Web log, allows the agency to communicate with the public in plain language about high interest or complex topics from a technical or regulatory standpoint. The NRC posts several blogs each week 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. As of November 2015, the agency published more than 600 blog posts since its launch in January 2011. Those posts have been viewed more than 751,000 times. The NRC received and posted more than 5,735 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, launched in August 2011, offers an opportunity for the agency to push out relevant information quickly in a simplified format. For example, the 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. As of November 2015, the NRC had more than 6,800 Twitter followers and continues to add new followers daily. The agency sent a total of 2,092 tweets over 54 months, for an average of 38 per month. The NRC's tweets sent during Hurricane Sandy in October 2012 were retweeted 130 times, potentially reaching more than 210,000 Twitter users.

The NRC's YouTube channel and Flickr photo gallery 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. Since launching the YouTube channel in August 2011, the agency posted about 140 videos, which have received nearly 104,000 views. More than 700 users subscribe to the NRC YouTube channel and are notified each time new content is posted. Since February 2012, the NRC has published about 2,200 photos and graphics to its Flickr account which have been collectively viewed more than 1.2 million times.

The agency launched Facebook in August 2014, rounding out its social media program. Since that time, its page has gained more than 2,200 likes. With nearly 270 posts, more than 22,000 people have engaged with NRC content on Facebook.

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 over 3,000 people from more than 30 countries, including members of Congress, nuclear industry representatives, international counterparts, and other stakeholders. The conference features presentations by the 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 Web streams 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 of this report.

The 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 U.S. law, through legislation enacted by the U.S. Congress, ensures the effective separation between the functions of the regulatory body and those of any other body concerned with the promotion or utilization of nuclear energy, as well as the independence of the regulatory body in making its safety-related decisions. Originally, the regulatory and promotional responsibilities for nuclear energy were combined into a single U.S. agency – the Atomic Energy Commission. In 1974, the U.S. Congress, through the Energy Reorganization Act of 1974, abolished the Atomic Energy Commission and divided its functions between two new agencies, the NRC and ERDA, which was succeeded by DOE in 1977. Section 201 of the Energy Reorganization Act of 1974, established the NRC as an "independent regulatory commission," while ERDA, now DOE, was established as a cabinet level agency of the U.S. President.

Congress conferred upon the NRC the licensing, inspection, and enforcement regulatory responsibility for all civilian uses of nuclear energy and materials as provided for in the Atomic Energy Act of 1954, as amended. The promotional and technology development functions were transferred under the Act to ERDA. This division resulted in the complete separation of regulatory from promotional responsibilities.

Given the NRC's status as an independent regulatory agency, the NRC's Commissioners, in contrast to the heads of cabinet level agencies like DOE, may be removed by the U.S. President only for "inefficiency, neglect of duty, or malfeasance in office." The NRC's independence allows it to promulgate regulations governing commercial nuclear power uses without submitting them to the cabinet-level OMB for review and approval.

Accordingly, the NRC has 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 commercial nuclear fuel cycle materials and facilities. The NRC is also 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.

The enactment of the DOE Organization Act in 1977 subsequently 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 nuclear-related activities to include nonproliferation and the environmental cleanup of contaminated DOE and certain other legacy 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 Ethics Rules Applying to NRC Employees and Former Employees

NRC employees must comply with governmentwide ethics rules contained in federal statutes and regulations. These rules are intended to ensure that every citizen can have confidence in the integrity of the Federal Government. The rules set standards that federal employees must follow in situations that may raise ethics concerns. For example, the rules restrict an employee's ability to accept gifts from regulated entities; prohibit the employee from working on matters in which he may have a conflicting financial interest, such as a matter involving a recent former employer; and preclude the employee from using his public position for private gain.

In addition to these government-wide rules, the NRC has two of its own ethics rules. The first rule establishes a Prohibited Securities List consisting of power reactor licensees and certain other entities that may be affected by the NRC's regulatory actions. NRC employees in designated positions cannot own stock in any company appearing on the Prohibited Securities List. The second rule requires that NRC employees obtain prior approval before accepting outside employment with certain types of employers, including any organization that operates in the commercial nuclear field.

When an NRC employee leaves the agency, he must comply with federal post-employment rules. These rules prevent the former employee from representing a non-federal party before the federal government on certain matters. The length of the restriction depends on the former employee's position while at the NRC and the extent of his participation in the matter on which he seeks to represent the non-federal party.

In addition to these rules, political appointees at the NRC must currently follow the commitments listed in the Ethics Pledge that President Obama issued in 2009. The Ethics Pledge further limits an appointee's ability to accept gifts, extends certain postemployment restrictions, and imposes additional requirements.

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 U.S. NRC, through the Atomic Energy Act, ensures that the primary responsibility for the safety of a nuclear installation rests with the licensee. The overall responsibility of licensees for ensuring safety of their facilities did not change as a result of the Fukushima accident. U.S. licensees continue to respond to new NRC regulatory requirements and initiatives that confirm and ensure that adequate measures to protect public health and safety are taken considering the lessons learned following the accident, as described in Sections 1.3.1 and 1.3.3 of this report.

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 for maintaining resources for managing accidents.

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 production and utilization facilities for commercial or industrial purposes, which includes nuclear power reactors. 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.

This responsibility for safety is implemented, in part, by having trained and qualified operators. Section 50.54 of 10 CFR, "Conditions of Licenses," identifies requirements that are conditions in every nuclear power reactor operating license. This regulation, in part, specifies the minimum requirements per shift for onsite staffing of the control room by operators and senior operators, including multiunit sites and shared control rooms (10 CFR 50.54(i) through (m)). Additionally, 10 CFR 50.120, "Additional Standards for Licenses, Certifications, and Regulatory Approvals," requires that each licensee establish, implement, and maintain a training program. The training program must incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. The training program must be developed to be in compliance with the facility license, including all technical specifications and applicable regulations. For additional information, see Sections 11.2, 12.3.2, and 12.3.3 of this report. The training program must be periodically evaluated and revised as appropriate to reflect industry experience as well as changes to the facility, procedures, regulations, and quality assurance requirements. Additional information on licensee training and accreditation programs is provided in Part 3 of this report.

Furthermore, licensees have financial responsibilities in the event of an accident. 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 license condition in 10 CFR 50.54(w) requires that licensees obtain insurance in an equivalent amount of protection covering the licensee's obligation, in the event of an accident at the licensee's reactor, to stabilize and decontaminate the reactor and the reactor site. Licensees are required to report the current levels of insurance or financial security and the sources of the insurance or security to the NRC on April 1 of each year. Additionally, licensees are required to have and maintain financial protection liability insurance for claims arising from accidents. Additional information on liability insurance can be found in Section 11.1.3 of this report.

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 February 4, 2015, 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.[11] 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

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.

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11 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.

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 (i.e., a Severity Level III or higher notice of violation or a greater-than-green Reactor Oversight Process finding) 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 also offered as an option for nonwillful (traditional) enforcement cases with the potential for civil penalties. 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. A base civil penalty has been established for each escalated severity level violation and for each type of licensee. Specific Commission approval is required for proposals to impose a civil penalty for a single violation or problem that is greater than three times the Severity Level I civil penalty value for that type of licensee.

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.

In the last 3 calendar years, the NRC issued the following significant enforcement actions to operating power reactors:

	Calendar Year		
	2013	**2014**	**2015**
Notices of violation without civil penalties	26	30	28
Civil penalties	1	1	3
Orders without civil penalties	8	7	2
Total enforcement actions	**35**	**38**	**33**

9.4 Openness and Transparency

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 of this report. 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 during an incident. 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.

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.

Policies of the U.S. NRC 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.

The NRC has not made specific changes to the priority of its safety programs addressed under this article as a result of the Fukushima accident. However, the actions that the NRC has taken are broadly supportive of the due priority to safety and a strong safety culture. For example, as part of the larger effort discussed in Sections 1.3.1 and 1.3.3, the NRC has ensured that licensees have adequate staffing to respond to emergencies and clearly defined roles and responsibilities for those responders. Similarly, the NRC requested that nuclear power plant licensees reevaluate potential seismic and flooding hazards and perform a mitigation strategies assessment of the reevaluated hazards.

10.1 Background

The NRC has a longstanding goal to move toward more risk-informed and performance-based approaches in its regulatory programs. In SRM-SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999, the Commission approved defining the terminology and expectations for evaluating and implementing initiatives related to risk-informed, performance-based approaches. A risk-informed approach is an approach whereby risk results and insights from a PRA that addresses a broad range of plant conditions are used, in a complementary manner with the traditional (deterministic) engineering concepts of defense-in-depth and safety margin, to establish requirements. In contrast, a solely deterministic approach would only address a limited number of design basis conditions and relies on conservatisms in the analyses. The risk-informed approach better focuses licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. A performance-based approach is an approach that establishes measurable (or calculable) outcomes to be met, instead of using prescriptive requirements that specify particular features, actions, or programmatic elements to be included in the design or process. Therefore, the performance-based approach provides more flexibility to the licensee in meeting the design or process objective. Implemented together, the risk-informed, performance-based approach uses risk insights, engineering analyses, judgment, the principles of defense-in-depth and safety margins, and performance history to:

- focus attention and resources on the most important activities and issues
- establish objective criteria for evaluating performance
- develop measurable or calculable parameters for monitoring system and licensee performance
- provide flexibility to determine how to meet the established performance criteria in a way that encourages and reward improved outcomes

- focus on the results as the primary basis for regulatory decisionmaking

The United States has made progress in developing and using risk information to give due priority to nuclear safety and establishing policies, programs, and practices that apply to licensee safety culture.

The NRC has developed extensive guidance on the role of PRA in U.S. regulatory programs and applies risk insights gained from PRAs to complement traditional engineering analyses. The increased use of risk information has improved issue-specific safety regulation and risk information has been used to evaluate proposed changes to the current licensing bases for individual plants. For example, 10 CFR, Section 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” allows licensees to use a risk-informed approach to categorize SSCs, and assign special treatment requirements, 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 continues to evaluate ways that risk insights can be used to enhance its regulatory framework in a risk-informed, performance-based manner.

On June 14, 2011, the NRC issued its “Final Safety Culture Policy Statement” and identified traits of a positive safety culture. All U.S. nuclear power plants have committed to conducting a safety culture self-assessment every 2 years and have committed to conducting monitoring panels as described in Nuclear Energy Institute (NEI) 09-07, “Fostering a Healthy Nuclear Safety Culture,” dated March 2014.

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 “Use of Probabilistic Risk Assessment Methods in Nuclear Activities; Final Policy Statement,” dated August 16, 1995.

10.3 Applications of Probabilistic Risk Assessment

The NRC uses risk insights gained from PRA in conjunction with traditional engineering analyses to resolve emergent issues, evaluate new and existing requirements and programs, and to provide a technical basis for risk-informed regulation. In addition, the NRC conducts research to improve data and methods used in 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. To assess the technical adequacy of the supporting PRA for risk-informed applications, 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 in 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 in December 2015.

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, 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: http://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp.html.

The NRC and industry representatives have cooperated in several areas and piloted programs to develop and apply risk-informed methodologies for specific regulatory applications. The staff uses the lessons learned from these activities to develop and publish detailed implementation 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 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. This PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist 1 year before the scheduled date for initial loading of fuel into the reactor. Each holder of a combined license must maintain and update the PRA every 4 years with upgraded consensus standards in effect 1 year prior to each required upgrade 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 applications of risk-informed inservice testing, of generally limited scope. Special treatment requirements for SSCs go beyond industry-established requirements for equipment classified as commercial grade. Special treatment requirements provide increased assurance that the SSCs are capable of meeting their functional requirements under design basis conditions. These special treatment requirements include additional design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements. In August 2001, the staff granted the licensee of the South Texas Project a risk-informed exemption request, which included an exemption from the prescriptive inservice testing requirements, regarding special treatment requirements for low-risk and nonrisk-significant safety-related nuclear components. Having successfully implemented this exemption at the South Texas Project, the staff developed a new rule, 10 CFR 50.69, to allow the application of risk insights to assign the special treatment requirements in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," for SSCs according to their safety significance.

The Commission approved the final rule 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 July 2006, proposed a categorization process to support implementation of 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, because the NRC does not review specific treatment plans as part of a licensee’s application to implement 10 CFR 50.69.

On December 17, 2014, the NRC issued a license amendment approving the Vogtle Electric Generating Plant, Units 1 and 2, pilot application of 10 CFR 50.69. As necessary, the lessons learned from this pilot will be incorporated into future revisions of the industry guidance and the NRC’s regulatory 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, Chapter 3 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 the Electric Power Research Institute (EPRI), for alternatives to the ASME Boiler and BPV Code, Section XI, Inservice Inspection Program.

ASME has also developed Code Case N-716-1, “Alternative Piping Classification and Examination Requirements, Section XI, Division 1.” Code Case N-716-1 is founded, in large part, on the risk-informed inservice inspection process as described in NRC-approved EPRI Topical Report 112657, “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” Revision B-A, issued in December 1999. NRC-approved EPRI Topical Report TR-1021467-A, “Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs,” dated December 2008, provides additional guidance to ensure that the PRA quality is sufficient to support a risk-informed inservice inspection program. Code Cases provide alternatives to existing ASME BPV Code requirements that ASME has developed and approved. RG 1.147, “Inservice Inspection Code Case

Acceptability, ASME Section XI, Division 1," Revision 17, issued in August 2014, identifies the Code Cases that the NRC has determined to be acceptable alternatives to applicable parts of the ASME BPV Code, Section XI. RG 1.147 endorses Code Case N-716-1.

The NRC regularly participates in the ASME BPV 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 probabilistic safety assessment (currently referred to as PRA by the NRC) 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 uses RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1, and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, both issued in May 2011, as guidance to improve plant technical specifications. Guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, Chapter 19, Section 19.2, "Review of Risk-Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance." Guidance on evaluating PRA technical adequacy is provided in NUREG-0800, Chapter 19, Section 19.1, Revision 3. More specific guidance related to risk-informed technical specification changes is provided in NUREG-0800, Chapter 16, Section 16.1, "Risk-informed Decision-making: Technical Specifications," Revision 1, dated March 2007, which includes changes to surveillance frequencies and completion times as part of risk-informed decisionmaking.

The industry and the NRC continue to increase the use of PRA in developing improvements to technical specifications. As discussed in a letter from NEI to the NRC dated June 8, 2001 (ADAMS Accession No. ML011690233), the industry proposed eight initiatives to improve existing technical specification configuration control requirements through the use of risk insights. The NRC worked with the industry Technical Specifications Task Force (TSTF) to develop and approve technical specification "change travelers" associated with seven of the proposed initiatives. The approved initiatives allow licensees to: (1) modify end states for some technical specifications required actions to allow certain equipment to be repaired during hot shutdown instead of cold shutdown, (2) eliminate shutdown requirements for unintentionally missed surveillances, (3) increase mode change flexibility, and (4) permit 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 (i.e., snubbers and other hazard barriers). The NRC also approved up to a 24-hour completion time for a very limited scope of technical specification systems (e.g., pressurizer heaters) when both safety trains are inoperable for Combustion Engineering plants. Additionally, the two initiatives considered to be the most significant expansion of the use of risk information for improving technical specifications are summarized below.

<u>Initiative 4b, "Risk-Informed Completion Times."</u> Initiative 4b modifies technical specification completion times, in accordance with NRC-approved topical report NEI 06-09, "Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, issued in November 2006, to reflect a

configuration risk-management approach that is more consistent with the 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Industry guidance and the South Texas Project pilot were approved in 2007. The associated technical specification change traveler can be found in TSTF-505, “Provide Risk-Informed Extended Completion Times—RITSTF [Risk-Informed Technical Specifications Task Force] Initiative 4B,” Revision 1. The NRC received a second pilot application in September 2012. The NRC staff is nearing completion of its review of the application, and is actively working to resolve any remaining technical issues and provide clarifying guidance. Four additional applications to implement this program for currently operating reactors have been received and are currently being reviewed by the technical staff. This program is expected to be widely adopted by licensees.

Initiative 5b, “Risk-informed Method for Control of Surveillance Frequencies.” Initiative 5b relocates most periodic frequencies of technical specification surveillances to a licensee controlled program in accordance with NRC-approved topical report NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1, dated April 2007. The associated technical specification change traveler can be found in TSTF-425, “Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b,” Revision 3, dated March 18, 2009. This program has already been implemented by more than half of the U.S. licensees and the NRC continues to receive and review applications for this initiative.

10.3.4 Development of Standards

The NRC worked with ASME and the American Nuclear Society (ANS) to update the 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.

Continuing to work with ASME and ANS, the agency is in the process of supporting issuance of the next revision to the PRA standard, which will be accompanied by Revision 3 of RG 1.200. The next revision of the PRA standard will include low power and shutdown modes, Level 2 and Level 3 PRA, and advanced light water and nonlight water reactor designs, in addition to updates to internal and external events portions, as needed.[12]

10.3.5 Level 3 Probabilistic Risk Assessment Project

As directed in Staff Requirements Memorandum (SRM)-SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities,” dated September 21, 2011, the staff is conducting a full-scope site Level 3 PRA that addresses all internal and external hazards, plant operating modes, reactor units, spent fuel pools, and dry cask storage.

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12 While an intermediate version, ASME/ANS RA-Sb-2013, was issued in September 2013, this version was not endorsed nor has it been used in place of ASME/ANS-RA-Sa-2009, which remains in effect today until the cited next revision is endorsed via RG 1.200, Revision 3.

The full-scope site Level 3 PRA project includes the following objectives:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data, that (1) reflects technical advances since completion of the NUREG-1150 study, titled, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated December 1990, and (2) addresses scope considerations that were not previously considered (e.g., shutdown and low-power operations, multiunit risk, and spent fuel storage).

- Extract new risk insights to enhance regulatory decisionmaking and help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

- Enhance and improve the PRA staff's capability and expertise, and documentation to make PRA information more accessible, retrievable, and understandable.

- Obtain insight into the technical feasibility and cost of developing new Level 3 PRAs.

Consistent with the objectives of this project, the Level 3 PRA study is largely being carried out using current PRA state-of-practice methods, tools, and data. However, there are several gaps in PRA technology, along with other challenges, that require advancement in the PRA state-of-practice. To address these gaps and challenges for the Level 3 PRA study, the general approach is to rely primarily on existing research and the collective expertise of the NRC's senior technical advisors and contractors, with limited new research for a few specific technical areas (e.g., multiunit risk).

Based on a set of site selection criteria and support from NEI, Vogtle Electric Generating Plant, Units 1 and 2, were selected as the volunteer site for the Level 3 PRA study. To enhance the efficiency in performing the study, the Level 3 PRA project team is leveraging the existing and available information on the Vogtle facility, the licensee's PRAs, and related research efforts.

The Level 3 PRA project team is using the following NRC tools for performing the Level 3 PRA study:

- Systems Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE)
- MELCOR Severe Accident Analysis Code
- MELCOR Accident Consequence Code System

In addition, the Level 3 PRA study is being made consistent with many of the modeling conventions used for the standardized plant analysis risk models, which are plant-specific PRA models used by the staff to support risk-informed regulatory activities.

10.4 Safety Culture

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

10.4.1 Safety Culture Policy Statement

Industry experience has shown the value of establishing and maintaining a positive safety culture. The NRC's Safety Culture 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. Safety and security are the primary pillars of the NRC's regulatory mission and consideration of both is an underlying principle of the Safety Culture Policy Statement.

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

- Respectful work environment—trust and respect permeate the organization

- Questioning attitude—individuals avoid complacency and continuously challenge existing conditions and activities in order to identify discrepancies that might result in error or inappropriate action

After publication of the policy statement, the NRC engaged the Institute of Nuclear Power Operations, NEI, and external stakeholders in the reactor community to develop a common safety culture language using the NRC's Safety Culture Policy Statement's traits as a basis. This language, which was finalized in early 2013, better aligns the industry's previous safety culture language with the NRC's previous safety culture language to allow for more clarification and enhance understanding of licensee performance. A 10th safety culture trait, "Decisionmaking—decisions that support or affect nuclear safety are systematic, rigorous, and thorough," was added in this common language effort for the reactor community. The NRC updated all guidance and inspection documents appropriately with the new common safety culture language and published NUREG-2165, "Safety Culture Common Language," in March 2014.

10.4.2 NRC Monitoring of Licensee Safety Culture

10.4.2.1 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.

10.4.2.2 Enhanced Reactor Oversight Process

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.

The Reactor Oversight Process applies the safety culture traits and attributes of NUREG-2165 to the inspection and assessment of licensee performance as described in Inspection Manual Chapter 0310, “Aspects within the Cross-Cutting Areas,” dated February 23, 2010. 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 safety culture (e.g., a cross-cutting aspect) is a significant causal factor of the finding. The NRC communicates the inspection findings to the licensee along with the associated cross-cutting aspect.

When performing inspections using IP 71152 “Problem Identification and Resolution,” dated February 26, 2015, inspectors have the option to review licensee self-assessments of safety culture. This IP also instructs inspectors to be aware of safety culture attributes 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 17, 2015, directs inspection teams to consider contributing causes related to the safety culture attributes as part of their efforts to fully understand the circumstances surrounding an event and its probable cause(s).

As part of the assessment process, 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 attributes.

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 attribute contributed to the risk-significant performance issues. If, through the conduct of supplemental inspection using IP 95002, "Inspection for One Degraded Cornerstone or any Three White Inputs in a Strategic Performance Area," dated February 9, 2011, the NRC determines that the licensee did not recognize that existing or suspected safety culture attributes 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 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," that was substantially revised in December 2015, 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. These activities range from requesting the licensee perform a safety culture self-assessment to a meeting between senior NRC managers and a licensee's Board of Directors to discuss licensee performance issues and actions to address persistent and continuing safety culture cross-cutting issues.

10.4.3 The NRC Safety Culture

The NRC fosters a culture in which all employees may live the NRC's values, demonstrate a positive safety culture, and adhere to the Principles of Good Regulation to support the NRC's mission to protect public health, safety, and the environment. The NRC culture includes a system of shared values, beliefs, and behaviors that demonstrate our collective commitment to emphasize safety as the overriding priority in our regulatory decisionmaking, and that recognizes the important role each employee plays in the NRC's success. The NRC is committed to creating and sustaining a positive work environment to ensure we remain a model regulator.

The NRC acknowledges the nature and purpose of a regulatory body is distinct from that of its licensees; therefore, the practical applications of ensuring a positive safety culture are slightly different and unique in some sense. Although many similarities regarding safety culture exist in any organization, the NRC emphasizes and relays the importance of safety culture as an inherent component of the broader NRC organizational culture that is complementary to, but distinct from, the NRC's regulatory oversight of licensees' safety culture.

The NRC emphasizes the notion that safety is every employee's responsibility. When each NRC employee demonstrates a level of responsibility for their behaviors and attitudes, which support a positive safety culture, it produces immeasurable gains that lead to higher operating margins across the board. Previous studies conducted at the NRC have revealed that high levels of key safety culture indices result in an engaged, enabled, and energized workforce– all of which comprise sustainable engagement. Thus, when safety culture indices increase, employee engagement increases. For this reason, the NRC has focused on achieving a positive safety culture and considers it to be a key driver of sustainable engagement.

Three key components of the NRC's safety culture include:

(1) Creating an environment that encourages all NRC employees and contractors to raise concerns and differing views promptly, without fear of reprisal. The free and open exchange of views or ideas conducted in a nonthreatening environment provides the ideal forum where concerns and alternative views can be considered and addressed in an efficient and timely manner that improves decisionmaking and supports the agency's safety and security mission.

(2) The NRC's commitment to the free and open discussion of professional views is illustrated by providing multiple ways for employees and contractors to raise mission-related concerns and differing views. Although all NRC employees and contractors are expected to discuss their views and concerns with their immediate supervisors on a regular, ongoing basis, there are times when informal discussions are not sufficient to resolve issues. The NRC uses a three-tiered approach for addressing concerns and differing views, including the processes described in Management Directive (MD) 10.160, "Open Door Policy," dated October 26, 2015, MD 10.158, "NRC Non-Concurrence Process," dated March 14, 2014, and MD 10.159, "The NRC Differing Professional Opinions Program," dated May 16, 2004. These directives provide increasing levels of formality to air differences: the broad Open Door Policy is least formal and does not require documentation, the Non-Concurrence Process requires documentation, and the Differing Professional Opinions Program is most formal and provides for a high level of agency review. The NRC believes that the existence of multiple channels for expressing disagreement helps create a positive environment for raising concerns by reducing barriers to expressing differing opinions. The Non-Concurrence Process and Differing Professional Opinions Program also support our openness value, in that when the process is complete, an employee can request to make the records public.

(3) The NRC conducts assessments of our safety culture and continually reviews results and develops action plans to improve. In addition, the NRC recognizes the need for continuous improvement to maintain a positive safety culture. Complacency lends itself to a degradation in safety culture when new information and historical lessons are not processed and used to enhance the NRC and its regulatory products.

The agency uses the Office of the Inspector General's triennial Safety Culture and Climate Survey, as well as postsurvey assessment activities (e.g., focus groups, and employee interviews), to assess the effectiveness of 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. The Office of the Inspector General has conducted the Safety Culture and Climate Surveys six times —in 1998, 2002, 2005, 2009, 2012, and 2015.

The Government-administered Federal Employee Viewpoint Survey provides an annual check on topics such as leadership, employee engagement and job satisfaction. The U.S. Office of Personnel Management has conducted the Federal Employee Viewpoint Survey since 2002 and annually since 2010. A survey such as this makes it possible to compare results over time to assess increasing or declining trends. Action plans are developed at the agency, office, and Region levels to address areas needing improvement, and those plans are evaluated each year and updated, as necessary.

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 are closely intertwined, and it is critical that consideration be given to the integration of safety and security activities so as not to diminish or adversely affect either. Although 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 the potential for adverse impact, the NRC maintains its attention to the interfaces between safety and security during both normal (day-to-day operations) and emergency conditions.

The NRC's mission statement and strategic goals are achieved, in part, through a regulatory framework that stresses the importance of maintaining both safety and security under all site conditions. The NRC continues its efforts in the areas of rulemaking, licensing, emergency planning, and inspection to recognize, establish, and improve this interface. For example, the NRC has been working multilaterally with the International Atomic Energy Agency and bilaterally with its international counterparts to promote this concept. In March 2009, the NRC also 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. In addition, the NRC issued RG 5.74, "Managing the Safety/Security Interface," in June 2009, describing acceptable methods that could be used to meet the safety and security interface requirements of 10 CFR 73.58. Revision 1 of RG 5.74 was issued in April 2015, to include cyber security as part of the safety and security assessment.

From 2004 to 2012, as part of the NRC's increased focus on security events after the events of September 11, 2001, security issues were considered in the Reactor Oversight Process through a different assessment process than safety issues. To enhance consistency in the assessment process, 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 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.

Satisfactory licensee performance in the Reactor Oversight Process cornerstones provides reasonable assurance of safe and secure facility operation during both normal and emergency conditions and assurance that the NRC's safety and security missions are being effectively accomplished. Like the other cornerstones, the security cornerstone contains IPs and performance indicators to ensure that its objectives are being met. The NRC evaluates safety and security interface issues relative to their implications among the cornerstones and in the cross-cutting areas of human performance, safety conscious work environment, emergency planning, and problem identification and resolution. Safety and security activities 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 safety and security interface processes 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.

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.

There have been no changes in licensee financial resource considerations as a result of the Fukushima nuclear accident. Training related to the orders and the proposed rulemaking related to Fukushima lessons learned are described in Sections 1.3.1 and 1.3.3 of this report.

11.1 Financial Resources

Currently, the financial qualification regulations of the U.S. NRC are codified in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." They require applicants for a construction permit, operating license, or combined license to provide reasonable assurance of adequate funds to safely construct and operate nuclear production and utilization facilities. This means that applicants must provide information specifying their legal and financial relationships with stakeholders, corporate affiliates, or financial institutions upon which the applicant is relying for financial assistance, and information to support the financial capability of each such entity to meet its financial commitment to the applicant. After closely examining the current financial qualification regulations, the NRC has determined that the details of these arrangements go well beyond the NRC's mandate of ensuring public health and safety. Therefore, the NRC is considering the conformance of the existing 10 CFR Part 50 standard to be consistent with a 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," standard requiring a licensee to only demonstrate that it "appears to be financially qualified" to construct and operate a facility safely.

Additionally, the NRC's regulations at 10 CFR 50.54(w) and 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," require licensees to maintain financial protection in the form of onsite and offsite liability insurance. This insurance provides the licensee with financial assistance for any claims of bodily injury and property damage resulting from a nuclear incident, and helps pay onsite recovery costs. Additional information can be found in Sections 11.1.3 and 11.1.4 of this report.

The NRC also maintains decommissioning funding and related reporting requirements under 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," and 10 CFR 50.82, "Termination of License," throughout the life of a reactor facility, and regularly reviews the status of licensees' decommissioning trust funds. These detailed reviews provide NRC reasonable assurance that licensees maintain adequate funds to safely decommission their facilities.

11.1.1 Financial Qualifications 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.

On April 24, 2014, the Commission issued SRM-SECY-13-0124, "Policy Options for Merchant (Non-Electric Utility) Plant Financial Qualifications," approving the staff's recommendation to conduct a rulemaking to amend the financial qualifications requirements in 10 CFR Part 50 to conform to standards contained in 10 CFR Part 70. The rulemaking would permit the inclusion of a license condition to assure applicant financial qualifications reflecting the revised standards for review, and require the applicant to submit a plan for how it will proceed to finance the construction and operation of the facility to ensure that the applicant has a well-articulated understanding of the size of the project it is undertaking and the financial capacity to obtain the necessary financing when the applicant is ready to start construction. The NRC staff is currently revising the draft regulatory basis for the rulemaking, based on public comments, and anticipates that the rulemaking activities will continue through calendar year 2017.

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, to 10 CFR Part 50, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits," provides more specific directions for evaluating the financial qualifications of applicants.

NUREG-1577, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1, provides staff guidance for its review and approval of an applicant's and licensee's financial qualification during initial plant construction and operations.

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 are also required to submit estimates of the total annual operating costs for each of the first 5 years of operation of their facilities including the sources of funds to cover these 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, including information about financial qualifications. Under the requirements in 10 CFR 50.33(f)(4), each application for a combined license submitted by a newly-formed entity organized for the primary purpose of constructing or operating a facility must include information showing: (1) the legal and financial relationships it has or proposes to have with its stockholders or owners, (2) the stockholders' or owners' financial ability to meet any contractual obligation to the entity that they have incurred or proposed to incur, and (3) any other information considered necessary by the Commission to enable it to determine the applicant's financial qualification.

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 and Federal Energy Regulatory Commission 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. On March 4, 2015, the NRC published OL/FR-ISG-2014-01, "Interim Staff Guidance - Reviewing and Assessing the Financial Condition of Operating Power Reactor Licensees, Including Requests for Additional Information," on its process for reviewing and assessing licensee financial conditions.

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 provides staff guidance for its review and approval of applicants' and licensees' financial qualifications during initial plant construction and operations, including license transfers. Specifically, NUREG-1577 requests staff to determine whether, in the case of a direct transfer, a proposed transferee is qualified to hold the license, or whether, in the case of an indirect transfer, the holder of the license is qualified to hold the license. The regulations at 10 CFR 50.80(b) require license transfer applicants to include as much of the information with respect to, among other things, the financial qualifications of the proposed holder of the license as required in section 10 CFR 50.33(f). The reviewer should evaluate the financial qualifications associated with these transfers by: (1) determining whether the proposed holder of the license will remain an electric utility following the direct or indirect transfer; (2) for nonelectric-utility applicants, reviewing the recent financial performance of the proposed transferee, or, if the proposed transferee is a new entity such as an operating, generating, or service company subsidiary, evaluating the ownership or participation agreement with its owners or other responsible party; and (3) identifying all parent companies that are not licensed by the NRC or did not undergo a 10 CFR 50.80 review.

11.1.2 Financial Qualifications Program for Decommissioning

The Atomic Energy Act establishes the basis for the NRC's regulations and guidance on decommissioning funding assurance. The NRC's regulations at 10 CFR 50.75 and 10 CFR 50.82 require an applicant or licensee to provide the NRC with reasonable assurance of its plan to safely decommission a facility, including a cost estimate, the mechanism (e.g., establishment of a dedicated trust fund) and schedule to pay for decommissioning, and a certification that financial assurance for decommissioning will be, or has been provided. Additionally, the NRC has a comprehensive decommissioning funding oversight program in place to provide reasonable assurance that sufficient funds will be available for radiological decommissioning of all U.S. commercial nuclear reactors. Under 10 CFR 50.75, this program requires operating reactor licensees to submit biennial Decommissioning Funding Status Reports, which includes, at a minimum:

- the amount of decommissioning funds estimated to be required pursuant to 10 CFR 50.75(b) and (c)
- the amount of decommissioning funds accumulated to the end of the calendar year preceding the date of the report
- a schedule of the annual amounts remaining to be collected
- the assumptions used regarding rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections
- any contracts upon which the licensee is relying
- any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report
- any material changes to trust agreements

For power reactors that have ceased operations and are in decommissioning, similar reports are submitted on annual basis under 10 CFR 50.82, and includes information regarding the amount of decommissioning funds spent over the calendar year and the amount of remaining funds needed to complete decommissioning.

NRC-required decommissioning trust funds are designed in such a way as to protect the funds from withdrawals for expenditures other than those specifically authorized by NRC regulations. The intent of the trust funds is to cover the costs associated with the radiological decommissioning of the reactor facility, resulting from the termination of the NRC-issued license.

11.1.3 Financial Protection Program for Liability Claims Arising from Incidents

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 incidents 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.

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 incident and (2) ensure that adequate funds are available to the public to satisfy liability claims if such an incident were to occur.

The Price-Anderson Act was most recently revised in 2005, when Congress renewed the Commission's authority to regulate insurance requirements for nuclear facilities until 2020. Under the current law, power reactors of 100,000 kilowatts electric or more must contribute to a funding pool that is enacted if the primary layer of financial protection (in the form of private liability insurance — now at $375 million) is exhausted. The NRC is required by Section 170(t) to adjust these amounts for inflation every 5 years based on the aggregate change in consumer price index.

Reactor operators must pay into a funding pool for the secondary layer of protection, called the "retrospective premium pool" in maximum annual installments not to exceed $18.963 million, up to a total of $121.255 million each. However, payment is required only if an incident exhausts 15 percent of the first layer of financial protection, that is currently set at $375 million, and only if and to the extent that, additional funds are needed to pay the damages. With 102[13] reactors currently participating in the system, the total financial protection available under the Price-Anderson Act for any one incident is approximately $13.3 billion (i.e., $375 million of primary coverage plus $121.255 million per reactor times 102 reactors), which is also the limit on liability. The limit of insurance coverage fluctuates as reactor licensees join or withdraw from the retrospective premium pool. A change in the limit also may occur when the amount of insurance coverage in either the primary or secondary tier is adjusted for inflation, as must be done every 5 years. In any potential incident, Congress will address any damages exceeding the total sum that reactor operators 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. Neither proof of fault, nor proof of what caused the incident, is necessary to issue a claim. This feature helps to ensure that potential claims are settled without delay from deliberation in the court system.

As of 2015, claims for more than 240 alleged incidents involving nuclear material have been filed under various liability policies since the inception of the Price-Anderson Act in 1957. To date, the insured losses and expenses paid are approximately $507 million. Insurance pools

13 The number or reactors participating in the Price-Anderson Act system depends on granted insurance exemptions for the respective reactors, and it is not dependent on the number of reactors currently in operation or reactors in decommissioning. In the U.S., there are currently 100 nuclear reactors in operation.

paid out a total of approximately $71 million in claims and litigation costs in association with the Three Mile Island incident.

Separate from the Price-Anderson Act, the U.S. is a party to the Convention on Supplementary Compensation for Nuclear Damage, which was developed under the auspices of the IAEA to be the basis for a global nuclear liability regime. The Convention on Supplementary Compensation for Nuclear Damage was adopted on September 12, 1997, and was opened for signature on September 29, 1997. The United States signed at that time. The United States deposited its instrument of ratification in May 2008. The Convention on Supplementary Compensation for Nuclear Damage entered into force on April 15, 2015. The current parties include Argentina, India, Japan, Montenegro, Morocco, Romania, the United Arab Emirates, and the United States.

11.1.4 Insurance Program for Onsite Property Damages Arising from Incidents

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). Onsite insurance provides financial protection to stabilize and decontaminate the reactor and reactor station site at which the reactor experiencing an incident is located.

The U.S. nuclear power industry has not experienced an incident involving offsite radioactive release within the scope of this program since the Three Mile Island, Unit 2, event in 1979.

11.2 Regulatory Requirements for Qualifying, Training, and Retraining Personnel

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, "Operator Licensing Examination Standards for Power Reactors," Revision 10, issued in December 2014. 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 in 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 and Licensed Operator Performance," dated September 24, 2014. 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 in 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 2013, 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.

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 program on human performance run by the U.S. NRC. This program has seven major areas: (1) human factors engineering, (2) emergency operating procedures and plant procedures, (3) staffing, (4) fitness for duty, (5) the 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 a number of licensees are replacing 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 a few industry requests to transfer operating licenses due to changes of ownership of nuclear power plants. Changes in ownership often involve changes in organizational structure. Some organizational changes may have the potential to affect human performance, especially if operations are prioritized in the structure over safety (e.g., safety organizations are marginalized or devalued when they report to senior management through an operational portion of the organization).

12.3 Significant Regulatory Activities

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

- human factors engineering
- emergency operating procedures and plant procedures
- 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 of this report describes training.

12.3.1 Human Factors Engineering

This section discusses human factors activities related to plant design.

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 in March 2007; NUREG-0700, “Human System Interface Design Review Guideline,” Revision 2, issued in May 2002; and NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 3, issued in 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 in September 2007, to review license amendment requests that credit the use of manual actions. Moreover, Information Notice 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.

To make some of the current human factors guidance simpler, clearer, and more relevant to the digital environment, the staff issued an interim staff guidance (ISG) DI&C-ISG-05, “Highly-Integrated Control Rooms—Human Factors Issues (HICR-HF),” Revision 1, dated November 3, 2008. This ISG addresses 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 of 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, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” The NRC plans to issue a regulatory guide that will endorse, in part, the 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.

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).

The NRC 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 of Review Standard (RS-001), "Review Standard for Extended Power Uprates," issued in December 2003. The agency recently reviewed and approved extended power uprates for Monticello Nuclear Generating Plant, Unit 1, in December 2013, and Peach Bottom Atomic Power Station, Units 2 and 3, in August 2014.

The NRC has also evaluated 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.

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 GL 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, "Domestic Licensing of Production and Utilization Facilities," 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. The NRC staff describes the violation and the overall event in "H.B. Robinson Steam Electric Plant – NRC Integrated Inspection Report 05000261/2010004 and 05000261/2010501: Assessment Follow-up Letter," dated November 12, 2010. A followup inspection 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. The findings are documented in Inspection Report 05000261/2011010, dated July 6, 2011.

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. Recommendation 8 was for the "strengthening and integration of emergency operating procedures, severe accident management guidelines [SAMGs], and extensive damage mitigation guidelines." Emergency operating procedures, SAMGs, and extensive damage mitigation guidelines had been developed in the United States at different times and for different purposes, without explicit requirements for their integration. In addition, SAMGs were implemented at U.S. nuclear power plants as part of an industry voluntary initiative and, subsequent to the Fukushima accident, the NRC found that their maintenance was inconsistent from site-to-site.

On November 13, 2015, the NRC published for public comment proposed requirements related to the mitigation of beyond-design-basis events. New strategies for the mitigation of beyond-design-basis external events (also known as diverse and flexible coping strategies or FLEX guidelines) are integrated with emergency operating procedures, such that they support an integrated accident response capability. The integrated response capability would include the capability to respond to beyond-design-basis accidents, such as the earthquake and tsunami that affected Fukushima. In addition, the proposed Mitigation of Beyond-Design-Basis Events rule would require that each licensee's integrated accident management response capability include sufficient staffing to support implementation of the strategies and guidelines in conjunction with the emergency operating procedures and a supporting organizational structure for directing and performing the strategies and guidelines. Additional information on the NRC Fukushima lessons learned activities can be found in Sections 1.3.1 and 1.3.3 of this report.

12.3.3 Shift Staffing

Governing Documents and Process. In 10 CFR 50.54(m), the NRC establishes minimum onsite staffing requirements for licensed operators and senior operators at 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 in 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," dated March 28, 2010, discusses appropriate requirements for operator staffing for small or multimodule (advanced reactor) facilities. The NRC's regulations do not currently address the possibility of more than two reactors being controlled from one control room; some applications are expected to include control of more than two modules in a single control room. In addition, small modular reactor designers have stated 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 reactor 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. The staff made revisions to its RG, as needed, to address control room staffing for small modular reactors in preparation for receipt of the NuScale small modular reactor design certification application.

Experience. As noted previously, NUREG-1791 provides guidance for the review of staffing exemption requests for new generations of advanced reactors, as well as the increased use of advanced, automated, and digital systems in existing plants. A key element is the review of the staffing plan validation, an evaluation using performance-based tests to determine whether the staffing plan meets performance requirements and acceptably supports safe operation. Recent license application review experience indicates that applicants may be challenged to establish simulation capabilities to support such validation activities while they are finalizing other aspects of the plant design.

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, "FFD Program Applicability to Categories of Individuals." This rule also applies to licensees and permit holders authorized to construct a nuclear power plant to cover personnel performing certain construction, management, security, and quality control activities. All fitness for duty programs must meet the following performance objectives: (1) provide 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 is managing the effects of fatigue on an individual's ability to safely and competently perform his or her duties.

On March 31, 2008, the NRC amended 10 CFR Part 26, Subpart I, "Managing Fatigue," to include requirements for the management of worker fatigue. Subpart I supersedes the Commission's "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors," documented in GL 82-12, "Nuclear Power Plant Staff Working Hours," dated June 15, 1982. It also 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. After implementation of the rule, the NRC received several petitions for rulemaking from members of the public, requesting the NRC to alleviate alleged impacts adverse to safety that were introduced when the rule was implemented. The petitioners asserted 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.

The NRC has issued reports on statistical data and lessons learned from licensee's fitness for duty program performance reports. The latest report is titled "Summary of Fitness-for-Duty Program Performance Reports for Calendar Year 2013," and can be located at http://www.nrc.gov/reactors/operating/ops-experience/fitness-for-duty-programs/performance-reports.html.

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. Currently, the database is being updated to include data with a safety culture perspective.

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,” 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.

IP 95003, “Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input,” provides the supplemental response for repetitive degraded cornerstones, multiple degraded cornerstones, multiple yellow inputs, or one red input to the NRC Assessment Action Matrix. IP 95003 was revised in February 2011, to include 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.

Subsequent inspections conducted in 2013, which are briefly discussed in the following paragraphs, continued to evaluate performance initiatives and safety performance at the sites.

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. The NRC determined that the commitments that Palo Verde previously made had been completed and decided to reduce its oversight at this site.

In 2013, human factors experts participated in IP 95003 inspection activities at Browns Ferry utilizing the guidance in the 2011 procedure. The overall result and conclusion of the inspection was that the plant was being operated safely and that the licensee had to aggressively continue the implementation of the licensee’s integrated improvement plan to achieve substantial performance improvement.

The NRC has decided to reduce its oversight at Browns Ferry and Palo Verde based on improvements made. The staff’s findings are documents in inspection reports that be found on the NRC’s public Web site:
http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/listofrpts_body.html.

Throughout the years, the NRC has continued to gather lessons learned in this area. Insights gained from the inspections were used as a basis to substantially update IP 95003 and Inspection Manual Chapter 0305 in December 2015. For example, IP 95003 provides more detailed guidance on conducting independent assessment of the licensee's safety culture. The revised procedure will be used during an inspection that will be conducted at the Pilgrim Nuclear Power Station in late 2016 or early 2017.

12.4 Fukushima Lessons Learned

There are human factors considerations to many of the Fukushima lessons learned, including three orders that were issued, which are described in Sections 1.3.1 and 1.3.3 of this report. For example, for the mitigating strategies order, licensees are required to validate that their strategies can be performed as described, including the operator actions. In addition, the proposed Mitigation of Beyond-Design-Basis Events rulemaking includes requirements for licensees to ensure that they can transition smoothly between the different types of response guidelines. Furthermore, human factors are considered as part of the RFI issued by the NRC in March 2012; specifically, the RFI required licensees to assess its emergency communications systems and staffing levels to ensure sufficient resources are available to respond to an accident involving all units at the site.

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, and regulatory guidance.

There have been no changes to the quality assurance regulatory guidance or licensees' quality assurance programs as a result of the Fukushima accident. However, continued compliance with existing programs and requirements is an important aspect of implementation of the lessons learned from Fukushima, which are further discussed in Sections 1.3.1 and 1.3.3 of this report.

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, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." 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, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” 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 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," Revision 1, issued in August 2015, 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 the 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 in February 1976, through RG 1.33, "Quality Assurance Program Requirements (Operations)," Revision 2, issued in February 1978, as complying with the requirements of Appendix B to 10 CFR Part 50. The NRC staff issued RG 1.33, "Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants," Revision 3, in June 2013, endorsing ANSI/ANS 3.2-2012, "Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants," dated March 20, 2012. ANSI/ANS 3.2-2012 is focused on quality assurance of plant operations because information on quality assurance of design and construction is contained in 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. 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 February 26, 2015, to verify 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 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 Vendor Inspection Program

The NRC interacts with manufacturers and suppliers of safety-related components through the NRC Vendor Inspection Program that inspects compliance with quality assurance and defect reporting requirements. Vendor inspections are conducted at vendor facilities 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, and to verify that the quality assurance program provides controls for reporting of defects and noncompliance. Inspection Manual Chapter 2507, "Vendor Inspections," dated October 3, 2013, provides guidance for these inspections.

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. This section explains verification of the physical state and operation of the nuclear installation by analysis, surveillance, testing, and inspection. Finally, this section discusses lessons learned from Fukushima and addresses the Vienna Declaration on Nuclear Safety, which was issued in February 2015.

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 Ensuring Safety Assessments throughout Plant Life

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.

Once a license is issued for a nuclear plant, the licensee must operate the plant in conformance with its license and its licensing basis. The licensing basis evolves throughout the term of the license because of the continuing regulatory activities of the NRC, as well as the activities of the licensee. The Commission engages in a large number of regulatory activities which, when considered together, constitute a regulatory process that provides ongoing assurance that the licensing bases of nuclear power plants provide an acceptable level of safety. Section 14.1.5 of this report discusses 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. Section 18.5 of this report demonstrates how the NRC continually evaluates new information, including lessons learned from operational experience and their potential impact on risk and overall plant safety.

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 safety 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 the 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, Section 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.

RG 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000, which endorses industry guidance document NEI 96-07, Revision 1, “Guidelines for 10 CFR 50.59 Evaluations,” dated February 2000, provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59.

On March 6, 2015, the NRC staff issued a report, “Review of Lessons Learned from the San Onofre Steam Generator Tube Degradation Event,” along with an accompanying White Paper, “10 CFR 50.59; the Process, Application to Substantial Modifications to Licensee Facilities, and NRC Staff Assessment of Licensee Implementation,” dated February 25, 2015. The San Onofre Nuclear Generating Station lessons learned report highlights important aspects of the guidance in NEI 96-07, Revision 1, related to issues with the San Onofre 10 CFR 50.59 screening and evaluation for the replacement steam generators. This was followed by the issuance of RIS 2016-03, “10 CFR 50.59 Issues Identified In NRC’s San Onofre Steam Generator Tube Degradation Lessons Learned Report,” issued in April 13, 2016.

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, "Domestic Licensing of Production and Utilization Facilities," or an early site permit, combined license, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 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. Section (e) of 10 CFR 50.71 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, or transformers.

Review Process, Regulatory Requirements, and Guidance Documents. Because uprates affect a reactor's licensed power level, licensees 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, the 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 in 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 in 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, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," 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. After the approval, the NRC performs inspections of the power uprate implementation using IP 71004, "Power Uprates," dated May 21, 2015, 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, within 12 months of application review acceptance for stretch power uprates, and within 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 May 2016, the NRC had approved 157 uprates, resulting in a gain of approximately 22,034 MWt (megawatts thermal) or 7,346 MWe (megawatts electric), at existing plants. The NRC is currently reviewing three power uprate applications that would authorize an additional 1,482 MWt. In addition, licensees plan to submit 10 measurement uncertainty recapture power uprate applications in the next 2 years. If these expected applications are approved, the resulting uprates would authorize an additional 571 MWt (190 MWe).

Peach Bottom, Units 2 and 3, Extended Power Uprate

On August 25, 2014, the NRC approved a 12.4 percent extended power uprate for Peach Bottom, Units 2 and 3. Stretch and measurement uncertainty power uprates were previously approved for Peach Bottom. As such, the extended power uprate represents a power level equivalent to 120 percent of the original licensed thermal power level. As part of the plant modifications associated with the extended power uprate, the steam dryer in each unit was replaced.

The NRC approval of the extended power uprate was based, in part, on the capability for the licensee to monitor, evaluate, and take prompt action in response to potential adverse flow effects as a result of extended power uprate operation on plant SSCs, including verifying the continued structural integrity of the replacement steam dryer. A license condition was added to the facility operating license for each unit, as part of the extended power uprate amendment, to provide the necessary requirements associated with potential adverse flow effects.

For Peach Bottom Unit 2, the licensee completed the plant modifications needed to implement the extended power uprate during the fall 2014 refueling outage, including installation of the replacement steam dryer. During the power ascension following the refueling outage, data collected at about the 89 percent power level (equivalent to the preuprate 100 percent power level) identified strain responses on the replacement steam dryer in the low frequency range that were not previously predicted by the extended power uprate approved methodology. As a result, the licensee needed to change the steam dryer stress analysis methodology to better account for the low frequency loads. After NRC review and approval of the methodology, Peach Bottom, Unit 2 reached the new 100 percent extended power uprate power level on May 15, 2015.

For Peach Bottom, Unit 3, the licensee completed the plant modifications during the fall 2015 refueling outage to implement the extended power uprate.

Monticello Nuclear Generating Plant Extended Power Uprate

On December 9, 2013, the NRC approved the extended power uprate for the Monticello Nuclear Generating Plant. The extended power uprate application was submitted in a letter dated November 5, 2008. The amendment authorized an increase of the maximum core thermal power level by approximately 13 percent, from the previously licensed thermal power level of 1,775 MWt to 2,004 MWt.

The extended power uprate represents a power level equivalent to 120 percent of the original licensed thermal power level. The Atomic Energy Commission (predecessor of the NRC) originally issued an operating license to Monticello for a thermal power level of 1,670 MWt. In September 1998, the NRC approved a 6.3 percent power uprate to increase the power output to 1,775 MWt.

As part of the plant modifications associated with the extended power uprate, the steam dryer in each unit was replaced. The original Monticello steam dryer was a parallel vane bank, square hood design by General Electric, which does not have perforated plates at the inlet and outlet sides of the vane banks. In 2011, the licensee replaced its original dryer with a Westinghouse steam dryer that consists of three parallel vane banks of octagonal shape and a cylindrical skirt.

The NRC approval of the extended power uprate was based, in part, on the capability for the licensee to monitor, evaluate, and take prompt action in response to potential adverse flow effects as a result of extended power uprate operation on plant SSCs, including verifying the continued structural integrity of the replacement steam dryer. A license condition was added to the facility operating license as part of the extended power uprate amendment to provide the necessary requirements associated with potential adverse flow effects. During the power ascension, data was collected at various power plateaus. Monticello reached the new 100 percent extended power uprate power level on July 1, 2015.

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 decision to seek license renewal rests entirely with the nuclear power plant owners and typically is based on the plant's economic situation and whether it can meet NRC requirements.

The NRC has established a license renewal process with requirements to ensure safe plant operation for up to 20 additional years at a time. The NRC's current schedule is to complete the review of a license renewal application 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 August 2016, eight license renewal applications, spanning 12 units, are under NRC review.

NRC's final rule on "Continued Storage of Spent Nuclear Fuel," supports the agency's licensing decisions, in particular new reactor licensing and reactor license renewal by allowing the NRC to proceed with environmental reviews of new reactors or license renewal applications without considering the site-specific effects of spent fuel storage after the end of the reactor's licensed life for operation in the environmental analysis. See Section 1.3.3 of this report for further details on Continued Storage.

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 provides requirements for maintenance and monitoring of active and passive SSCs.

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 in 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 in 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 in 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 developing the Standard Review Plan for license renewal and the Generic Aging Lessons Learned Report for plants intending to operate beyond 60 years. Additional information on plant operation beyond 60 years can be found in Section 14.1.4.3 of this 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 license renewal

environmental review requirements under 10 CFR Part 51 are founded on the conclusion that certain environmental issues can be resolved generically and do not need to be evaluated in each plant-specific review. 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.

In June 2013, the agency amended Part 51 and its technical basis documented in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," to incorporate lessons learned and knowledge gained from previous license renewal environmental reviews conducted since 1996. During the development of the revised rule, the NRC added new environmental impact issues and consolidated similar ones. No environmental issues were deleted. The NRC's revised Table B-1 and updated NUREG-1437 identify 78 environmental issues; of these, 59 issues are considered generic or applicable to all nuclear power plants, 17 issues require a plant-specific analysis, and 2 issues require further information and remain uncategorized. The NRC conducts independent reviews of these environmental impacts to determine whether the effects are significant enough to preclude license renewal as an option for energy-planning decisionmakers. In June 2013, the NRC also updated its associated guidance documentation for license renewal applicants and its technical guidance for use by NRC staff. RG 4.2, "Preparation of Environmental Reports for Nuclear Power Plants License Renewal Applications," Supplement 1, Revision 1, provides guidance to applicants preparing environmental reports to be included as part of license renewal applications. NUREG-1555, Supplement 1, "Standard Review Plans for Environmental Reviews of Nuclear Power Plants: Environmental Standard Review Plan for Operating License Renewal," Revision 1, guides the NRC staff's review of the environmental issues associated with license renewal.

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 August 2016, 83 reactors, including Kewaunee, Vermont Yankee, Pilgrim, Unit 1, and FitzPatrick,[14] have received renewed licenses. Thirty-nine of the 81 reactors have completed 40 years of operation and are operating in the extended period. One reactor entered the period of extended operation in 2015, and six reactor units are expected to enter the period of extended operation in 2016. On the basis of industry statements, the NRC expects that almost all remaining plants that have yet to tender license renewal application will apply for license renewal. Please refer to http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html for a list of the plants that are expected to apply for license renewal.

14.1.4.3 Operating beyond 60 Years

The provisions of 10 CFR Part 54 allow a previously renewed operating license to be subsequently renewed with no additional requirements imposed and no limit on the number of times a license can be subsequently renewed, provided that it is justified and that safety is ensured. 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

14 Subsequent to receiving their renewed licenses, Kewaunee, Vermont Yankee, Pilgrim Unit 1, and Fitzpatrick announced that they would cease operations.

license renewal once it enters the initial period of extended operation (i.e., the 20-year renewal period beyond its initial 40-year license period). Several industry representatives have expressed an interest in operating nuclear power plants beyond 60 years and on November 5, 2015, the Commission received a formal letter of intent to pursue such a renewal for Surry Power Station, Units 1 and 2 in 2019. The renewed operating licenses for Surry Power Station, Units 1 and 2 were issued on March 20, 2003, and will expire on May 25, 2032, and January 29, 2033, respectively. On June 7, 2016, the Commission received a formal letter of intent to pursue a second license renewal for Peach Bottom Atomic Power Station, Units 2 and 3, in 2018. The renewed operating licenses for Peach Bottom Atomic Power Station, Units 2 and 3, were issued on May 7, 2013, and will expire on August 8, 2033, and July 2, 2034, respectively.

To prepare for the review of subsequent license renewal applications, on January 31, 2014, the NRC staff submitted to the Commission SECY-14-0016, "Ongoing Staff Activities to Assess Regulatory Considerations for Power Reactor Subsequent License Renewal." In SRM-SECY-14-0016, dated August 29, 2014, the Commission affirmed that the current regulatory framework for the first license renewal (i.e., operation from 40 years to 60 years) is sufficient to support the review of subsequent license renewal. In addition, in SRM-SECY-14-0016 the Commission directed the staff to:

- Continue to update the license renewal guidance, as needed, to provide additional clarity on the implementation of the license renewal regulatory framework.

- Address Option 2 and Option 3 as presented in SECY-14-0016 through alternative vehicles (e.g., issuance of generic communications, voluntary industry initiatives, or updates to the Generic Aging Lessons Learned Report). Option 2 recommended minor editorial changes to 10 CFR Part 54 to add alternate fracture toughness requirements and clarify how existing recordkeeping requirements apply to newly identified systems, structures, and components. Option 3, which includes Option 2, recommended an expansion in scope of 10 CFR Part 54 to include equipment associated with 10 CFR 50.54(hh)(2) and adds a provision to address timely renewal so that a licensee must implement aging-management activities before the expiration of its current license.

- Submit an information paper to the Commission reporting on the progress of the implementation of the inspection enhancements described in the Reactor Oversight Process Enhancement Project related to aging management and the Inspection Procedure Operating Experience Update Process.

- Keep the Commission informed on the progress of technical issues resolution, research activities, the staff's readiness to receive and evaluate the acceptability of an application for subsequent license renewal, and any further need for regulatory process changes, or rulemaking, related to subsequent license renewal.

- Emphasize in communications with industry the need to strive for satisfactory resolution of these issues before the NRC begins a review of any subsequent license renewal application.

To address the unique aspects of material aging and degradation that apply to subsequent license renewal, the NRC is collaborating on research activities with both domestic and international partners to ensure that important research topics are being addressed and to effectively leverage both resources and knowledge.

The NRC, in cooperation with the U.S. DOE Light Water Reactor Sustainability Program, completed NUREG/CR-7153, “Expanded Materials Degradation Assessment (EMDA),” Volumes 1 through 5, dated October 2014, to identify the most significant technical issues for nuclear power reactor operations beyond 60 years. The expanded materials degradation assessment ranked the significance, current knowledge, and uncertainty associated with aging-related degradation issues that could affect systems, structures, and components over 80 years of operation. As outlined SRM-SECY-14-0016, the major technical issue areas are:

- reactor pressure vessel neutron embrittlement at high fluence
- irradiation assisted stress corrosion cracking of reactor internals and primary system components
- concrete and containment degradation
- electrical cable qualification and condition assessment

Licensees and applicants have the primary responsibility for providing the technical basis to support their safety analysis and application for a subsequent license renewal. NRC staff conducts confirmatory research to independently verify licensee data, determine safety margins, and explore uncertainties. The NRC continues to track industry’s work in this area, evaluate areas for research, gather data to help assess the effectiveness of licensee’s aging management programs, and provide confirmatory research on the results of industry’s work. Results from NRC’s research will be used, in part, to confirm the adequacy of industry’s technical basis for a subsequent license renewal and the associated aging management programs. The aging management programs are cornerstones for managing materials degradation in safety-significant components during a subsequent license renewal. In addition, the NRC research will support and increase the efficiency of staff review of subsequent license renewal applications.

To support the review of the first subsequent license renewal application in 2019, the NRC staff is developing guidance documents to address the unique aging management needs for subsequent license renewal using as a starting point the existing license renewal guidance documents, NUREG-1801, Revision 2, and NUREG-1800, Revision 2. The NRC staff plans to issue the Generic Aging Lessons Learned Report and the Standard Review Plan for operations beyond 60 years by mid-2017.

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 plant 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.

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 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 provides 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 the NRC-required changes in the licensing basis, a licensee may also voluntarily seek changes to the current licensing basis for its facility. These changes are subject to the NRC regulations 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 regulations 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

Decades ago, 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. Later, in 1989, the NRC extended backfitting-style protections to nuclear power plants licensed under the new regulatory approval processes in 10 CFR Part 52. These new processes are the early site permit, standard design certification, and combined license. The NRC also provided backfitting protection to major fuel cycle facilities, independent spent fuel storage installations, and gaseous diffusion plants.

In 1988, the NRC amended the Backfitting Rule to state that economic costs will not be considered through a backfit analysis 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 adequate protection and compliance backfits, respectively. For example, 10 CFR 50.109(a)(5) states that the Commission shall always require the backfitting of a facility if it determines that such regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security. For backfits other than adequate protection and 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. For standard design certifications, there are additional criteria that must be satisfied to permit backfitting.

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.[15] 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 because 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).

15 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. The Generic Issues Program is discussed in Section 6.3.6 of this report.

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.

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, an integrated plant assessment[16] 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. A risk-informed approach to regulatory decisionmaking considers risk insights together with other factors to establish requirements that focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety. 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 4 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 most significant issues and 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 reported by the licensee to measure and assess plant performance. Together, the performance indicators and inspection findings give the information needed to support continuous reviews and assessments of plant performance. The Reactor Oversight Process also features comprehensive quarterly reviews and expanded annual 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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16 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, offnormal 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 nuclear power plant licensees to maintain a quality assurance program. Quality assurance comprises all those planned and systematic actions necessary to provide 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 means to control their 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. The audits are performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Management reviews the audit results and appropriate followup action is initiated.

14.1.5.7 The NRC's Regulatory Process Compared with International Safety Reviews

The IAEA and the Western European Nuclear Regulators' Association (WENRA) have developed guidance[17] 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

17 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.

After the 2010 International Regulatory Review Service (IRRS) mission in the United States, the NRC undertook a limited-scope pilot effort and a supplemental evaluation to review a sample of periodic safety review summary reports from other regulators to identify areas that could potentially inform the NRC's regulatory processes. The NRC issued a report titled, "Findings from the Staff's Evaluation of Periodic Safety Reviews from Other Countries," dated April 24, 2015. Based on the pilot effort and the supplemental evaluation, the NRC staff concluded that it is reasonable to expect that the U.S. regulatory approach would be sufficient for detecting and correcting the plant-specific issues documented in the periodic safety review summary reports, if they were to occur in U.S. plants. Hence, changes to the existing regulatory processes were deemed unnecessary. Discussions of the findings from other countries' periodic safety reviews present a valuable opportunity for NRC to stay apprised of international experiences in assessing reactor safety. The NRC welcomes such discussions during bilateral and multilateral exchanges as appropriate. Detailed information on the IRRS mission report can be found in Section 8.1.5.2 of this report.

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 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. In addition, the NRC has instituted a series of security enhancements for nuclear power plants since the September 11, 2001, terrorist attacks. These enhancements include upgraded physical security plans, enhanced security officer training, increased security patrols, additional physical barriers, greater stand-off distances for vehicle checks, and more restrictive site access controls, among others. Sections 10.5, 16.6.4 and 18.3.2.3 of this report provide additional discussions on the NRC's response actions after September 11, 2001, in the areas of safety and security interface, and cyber security. Separately, the NRC has undertaken significant actions to enhance the safety of nuclear power reactors in the U.S. following the Fukushima accident in Japan on March 11, 2011. The Fukushima-related actions include, but are not limited to, required implementation of mitigation strategies to respond to beyond-design-basis events; ensuring severe accident capable hardened containment vents for boiling water reactors with Mark I and II containments; reevaluation of seismic and flooding hazards using present-day guidance, methods, and information; and enhancing spent fuel pool instrumentation. Sections 1.3.1, 1.3.3, 6.4, 14.3, and 16.9 of this report provide additional details on the Fukushima-related actions.

Second, the NRC and the U.S. nuclear industry have more than 30 years of experience 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, "Codes and Standards," the NRC enumerates requirements for applying industry codes and standards to nuclear power reactors during design, construction, and operation. For example, this section incorporates by reference Section III and Section XI of the ASME Operation and Maintenance Code for nuclear power plants.

Through analysis, surveillance, testing, and inspection, the 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, testing, and inspections and take corrective action.

Under special circumstances, the Commission may also require under 10 CFR 50.54(f) that licensees submit written statements to the Commission. If necessary to ensure safe operation, the Commission can 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

The events at Fukushima Dai-ichi highlighted the need for safety improvements for nuclear power plants related to beyond-design-basis natural hazards. As described in Section 1.3.1 and 1.3.3 of this report, the NRC issued orders and a request for information to its licensees in response to the accident, and initiated a significant rulemaking to further enhance safety. The NRC also became involved in a host of international activities related to safety assessment.

Immediately after the event, using the existing Reactor Oversight Process, the NRC conducted inspections and issued orders, INs, and bulletins to aid in determining the preparedness of U.S. nuclear power plants to withstand a similar event. Furthermore, the Reactor Oversight Process will be used to assess and verify that changes currently being implemented in response to lessons learned from the accident were completed properly.

14.4 Vienna Declaration on Nuclear Safety

The NRC carries out many regulatory activities that, when considered together, provide for a comprehensive and systematic assessment and review to ensure public health and safety. One of the agency's main programs is the Reactor Oversight Process, which includes the use of baseline, periodic, special inspections, and daily oversight. Throughout the program, the NRC inspects, monitors, and assesses safety performance, and solicits feedback. Sections 14.1.1 and 6.3.2 of this report provide more information on the use of the Reactor Oversight Process. Section 14.1.5 of this report includes more information on how the NRC activities, including inspections, audits, research, backfitting, license renewal, and probabilistic risk assessment, are used to ensure that nuclear power plant safety is being continually assessed and assured over the entire period of operation.

The NRC also recognizes that the effective use of lessons learned from domestic and international operating experience is important for protecting the health and safety of people and the environment. The NRC screens operating experience for safety significance and generic implications, including the need for further action, as delineated in Section 6.3.5 of this report. The NRC communicates information internally to ensure that technical staff is able to factor operating experience into their reviews of plant safety. The NRC staff maintains communications with INPO to ensure that relevant operating experience reviewed by the industry is also considered by NRC reviews. The NRC communicates in public forums through the issuance of generic communications to provide NRC operating experience insights to the industry, the public, and the international community. In addition, the staff can make revisions to IPs when operating experience reviews indicate potential areas of concern for safety that may be reviewed through the inspection program. Section 19.7 of this report provides more information about the operating experience program.

Of significance, in light of the Fukushima accident, the NRC has taken a large number of actions to strengthen the protection of U.S. nuclear plants against events that could exceed a plant's design basis. For example, the NRC issued regulatory requirements in the form of three orders. These orders are discussed in Sections 1.3.1 and 1.3.3 of this report. The U.S. nuclear industry has taken a number of actions in response to the accident as well. The U.S. industry actions are summarized in Part 3.

To a large extent, the international community conducts periodic safety reviews to assess operating experience, technical developments, and other aspects such as the cumulative effects of plant aging. In contrast, the NRC uses routine and ongoing comprehensive safety inspection, audit, and assessment programs that deal with specific safety issues, significant events, and changes in safety standards and practices as they arise. These programs, as applied by the NRC with the appropriate scope, frequency, depth, and rigor, achieve the same review standards and objectives as a periodic safety review. This was demonstrated by the NRC's response to Fukushima, which reflects the NRC's regulatory approach of promptly addressing new information when discovered and taking appropriate regulatory action in a timely fashion, rather than awaiting a periodic review.

The effectiveness of the NRC's regulatory approach was evaluated by the IAEA IRRS mission and followup mission conducted in the NRC in 2010 and 2014, respectively. During the 2010 IRRS mission, the NRC correlated its regulatory programs to the 14 periodic safety review safety factors to clearly demonstrate that the NRC programs robustly meet the intent of the periodic safety review. The IRRS team concluded that the NRC has a number of processes in place, including a robust and mature inspection program, that meet the intent of a periodic safety review and that ensure that licensed facilities are properly meeting regulatory requirements. The results of the IRRS mission and the alternate program that the United States employs in lieu of conducting periodic safety reviews are further discussed in Sections 8.1.5.2 and 14.1.5 of this report.

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, United States radiation control measures are founded on radiological risk assessments by the United Nations Scientific Committee on the Effects of Atomic Radiation and the United States 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 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 U.S. 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) there is an assumption regarding a direct and proportional relationship between radiation exposure and cancer risk with all radiation doses (known as the Linear-Non-Threshold model). 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. Exceeding these dose limits may result in the issuance of violations and other sanctions to ensure compliance and prevent recurrence. There is a lengthy history of the doses being kept within the limits for workers (NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 2013: Forty-Sixth Annual Report," Volume 35, issued in December 2015), and members of the public living near nuclear power plants (NUREG/CR-2907, "Radioactive Effluents from Nuclear Power Plants Annual Report 2009," Volume 15, issued in August 2013.) 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, “Domestic Licensing of Production and Utilization Facilities,” 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 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 a recommendation 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 Backfitting 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 Commission 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 initially 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. However, in light of comments and feedback received on contemplated changes to 10 CFR Part 20, the NRC staff is no longer developing a regulatory basis for the revision to this rule. The staff has determined that the current NRC regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment.

Additional information on the options to revise radiation protection regulations and guidance can be found in the NRC's public Web site at: http://www.nrc.gov/about-nrc/regulatory/rulemaking/potential-rulemaking/opt-revise.html.

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, staff 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, international standards, consensus national standards, and authoritative recommendations 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, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," 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 in 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(h), 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 public and occupational radiation protection from nuclear power plant operations 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, and fully implemented in 1994, adopted the recommendations, quantities, and models recommended in ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection," issued in 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 in 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 in 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 – General Safety Requirements," issued in November 2014. 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.

10 CFR Part 50. 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, “Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors”; Appendix I, and 10 CFR 50.34(b)(3) and (h), which require NRC review of the plant radiation sources and protection programs. In 10 CFR 50.36a, “Technical Specifications on Effluents from Nuclear Power Reactors,” 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, in effective dose equivalent, appear in 10 CFR 50.34(a)(1)(ii)(D) for evaluating design basis accidents associated with licensing actions that have been submitted to the NRC since 1997. (The pre-1997 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).) In light of comments and feedback received on the contemplated changes to 10 CFR Part 50, Appendix I, the NRC staff is no longer developing a regulatory basis for the revision to 10 CFR Part 50, Appendix I. The current NRC regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment.

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, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 3, issued in May 2000, 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 in June 1978.

The NRC maintains an active regulatory inspection program that includes routine baseline inspections and supplemental inspections, as needed. Significant health physics problems can trigger 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 type of maintenance performed strongly influence the doses, the individual plant collective doses fluctuate from year to year. Before the nuclear plant accident in 1979 at Three Mile Island, Unit 2, the average collective dose per reactor varied substantially. After the accident, the collective worker doses increased because 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, collective doses have declined steadily by more than a factor of 10, to the current level of 0.71 person-Sv (71 person-rem) per reactor. The average collective dose for each BWR in 2014 was 1.09 person-Sv (109 person-rem). The average collective dose for each PWR in 2014 was 0.51 person-Sv (51 rem). The collective dose in a BWR is approximately a factor of 2 higher than PWRs, in part because of the larger work force at BWRs.

In 2014, 174,851 workers at nuclear plants were monitored for radiation exposure. Of these, 70,844 workers received a collective measureable dose of 71.24 person-Sv (7,124 person-rem) for an average of 0.0010 Sv (0.10 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 20.1406, "Minimization of Contamination," 10 CFR 50.34a, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, define the ALARA objectives for effluents from nuclear power plants. The regulations in 10 CFR 20.1406 require the minimization of contamination by conducting operations to minimize the introduction of residual radioactivity into the site, including the subsurface. Appendix I to 10 CFR Part 50 and 10 CFR 50.36a also establish regulations on effluent monitoring, environmental monitoring, investigations, land-use censuses, and reporting. Section IV.B of Appendix I 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 for ALARA are complemented by 10 CFR Part 20.1501, "General," which requires, in part, that a licensee perform surveys, including the subsurface, 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.

The NRC staff continues to provide the public with current information on control of radiation exposure to members of the public on its Web site at http://www.nrc.gov/about-nrc/regulatory/rulemaking/potential-rulemaking/opt-revise.html. Information posted on the NRC Web site 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

The NRC has not made any changes to its radiation protection programs in light of lessons learned from the accident. Since the accident at Fukushima, there have been studies undertaken by the United Nations Scientific Committee on the Effects of Atomic Radiation, the World Health Organization, and the Fukushima Medical University to evaluate the health effects from the accident, particularly the potential impact of radioactive iodine on children's thyroids. To date, no study has detected an increase in thyroid nodules or cancer among the pediatric population because of the accident. However, the NRC will continue to monitor these studies to determine if any policy changes are necessary.

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 responsibilities of the U.S. NRC 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. After 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 Section 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 in 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 in November 1980, describe the emergency planning basis. NUREG-0654/FEMA-REP-1 is being revised to align with the NRC emergency preparedness rule changes, which became effective in December 2011, and with the revised Federal Emergency Management Agency (FEMA) Radiological Emergency Preparedness Program manual issued in 2011. The NRC significantly enhanced its emergency preparedness regulations to address lessons learned from the September 11, 2001, terrorist attacks and security events. This is further discussed in Section 16.6.4 of this report.

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, “Emergency Planning and Preparedness for Production and Utilization Facilities.” 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 because 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 8-year exercise cycle.

Through the Steering Committee for Emergency Planning, established under the NRC-FEMA 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)(4)). Appendix E (Section IV.C.1) of 10 CFR Part 50 defines four emergency classification levels in order of increasing severity: (1) notification of unusual events, (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.

Emergency action level development guidance was initially established in GL 79-50, “Emergency Plans Submittal Dates,” and was subsequently established in NUREG-0654/FEMA-REP-1, which was endorsed as an approach for the development of an emergency action levels scheme through RG 1.101, “Emergency Planning and Preparedness for Nuclear Power Reactors,” Revision 2, dated October 1981. NUREG-0654/FEMA-REP-1 defines and gives examples of initiating conditions for the four emergency classification levels. These conditions form the basis for each licensee to establish specific thresholds and indicators, known collectively as emergency action levels. These action levels reflect specific plant conditions (e.g., plant system status, inplant and effluent radiological parameters, and other inplant hazards) or external events (e.g., flooding, earthquakes, high winds, security events) for each of the four emergency classification levels.

As industry and regulatory experience was gained with the implementation and use of emergency action levels schemes, the industry issued revised emergency action levels scheme development guidance to reflect lessons learned. To date, NUMARC/NESP-007, “Methodology for Development of Emergency Action Levels," dated January 1992, and NEI 99-01, "Methodology for Development of Emergency Action Levels," Revisions 4, 5, and 6, were provided to the NRC for review and endorsement as generic (nonplant-specific) emergency action levels development guidance. RG 1.101, Revisions 3 and 4, endorsed NUMARC/NESP-007 and NEI 99-01, Revision 4, dated January 2003, as acceptable alternatives for licensees to consider in the development of their plant-specific emergency action levels schemes, and allowed licensees to develop plant-specific emergency action levels based upon an alternative approach. The NRC endorsed NEI 99-01, Revision 5, dated February 2008, through a letter dated February 22, 2008. The NRC endorsed NEI 99-01, Revision 6, through a letter dated March 28, 2013. Emergency action levels development guidance for the Advanced Passive (AP)1000 and the General Electric-Hitachi's Economic Simplified Boiling Water Reactor (ESBWR) reactor designs is provided in NEI 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, dated July 2009.

The emergency action level development guidance contained in GL 79-50, NUREG-0654/FEMA-REP-1, NUMARC/NESP-007, and NEI 99-01, Revisions 4, 5, and 6, are all considered generic emergency action level scheme development guidance, as they are not plant-specific and may not be entirely applicable for some reactor designs (note that NEI 07-01 is only applicable to the AP1000 and ESBWR designs). However, the guidance contained in these documents bounds the most typical accident or event scenarios for which emergency response is necessary, in a format that allows for industry standardization and consistent regulatory oversight. Most licensees choose to develop plant-specific emergency action level schemes using the latest NRC-endorsed guidance with appropriate plant-specific alterations, as applicable. Under 10 CFR Part 50, Appendix E, Section IV.B (2), a revision to an emergency action level must be approved by the NRC before implementation if the licensee is changing action level schemes.

Although not required under existing U.S. emergency preparedness regulations contained in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, several 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 operating 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 in November 1980).

The nuclear industry developed SAMGs in response to the Three Mile Island 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. Although not required under U.S. emergency preparedness regulations, SAMGs are intended for use by plant technical staff, usually 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, licensees to develop and implement SAMGs. Because SAMGs are voluntary, formal training and licensing of plant operators and emergency preparedness regulations do not require them to be addressed. Nevertheless, operating power reactor licensees committed to implementing SAMGs following guidance that was agreed upon with the NRC and completed that implementation by the end of 1998.

Following the Fukushima Dai-ichi accident, the nuclear industry and the NRC revisited the issue of SAMGs. Immediately following the accident, the NRC conducted inspections of the voluntarily-implemented SAMGs under TI 2515/184, “Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs).” The inspectors observed inconsistent implementation of SAMGs and attributed it to the voluntary nature of the initiative. As a result of this effort, the Commission once again considered whether SAMGs should be required of licensees, but concluded that the potential risk of a severe accident, which has diminished since the Three Mile Island accident through the imposition of additional pre-core damage safety requirements over the years, does not rise to the level for which it would be appropriate to require SAMGs. In parallel with this NRC reexamination of the need to require SAMGs, the nuclear industry made a number of changes to the technical basis document that supports development of the guidelines in order to appropriately treat the lessons learned from that event. These revisions were used by the owners groups to revise and update their generic guidelines, which in turn are being used by the industry to update the facility-specific guidelines. These improvements to the industry’s SAMG program, including revised commitments on the part of the nuclear industry for a process under which the SAMGs would be maintained and updated, contributed to the NRC’s conclusion that continued status as a voluntary program is appropriate. In addition, the NRC initiated changes to the Reactor Oversight Process to include inspection of the resulting programs as voluntarily-imposed standards.

Additionally, the NRC is in the process of developing regulatory guidance for the Mitigation of Beyond-Design-Basis Events Rulemaking that would provide for appropriate coordination of the voluntarily-maintained SAMGs with the guidance and strategies required by the rulemaking for beyond-design-basis events resulting from natural phenomena as well as the extensive damage mitigating guidelines required of the licensees after the terrorist events of September 11, 2001. On August 27, 2015, the Commission issued SRM-SECY-15-0065, “Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events,” directing the staff to update the Reactor Oversight Process to explicitly provide periodic oversight of industry’s voluntary implementation of the SAMGs. Because this is a longer-term activity, the NRC staff envisions that the Reactor Oversight Process and relevant inspection procedures will be updated by December 31, 2020, to allow for oversight of the site-specific voluntary incorporation of SAMGs in generic guidance revisions. The boiling-water reactor owners group and the pressurized-water reactor owners group have also issued revised versions of their generic guidance for SAMGs based on the industry’s technical basis document update and continue to improve this guidance to reflect lessons learned in the implementation.

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, issued in November 2011, and EPA 400-R-92-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,” issued in May 1992, which is 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 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 in 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's 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 May 2013 and the update of its associated annexes. Additionally, the U.S. Department of Homeland Security (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 was revamped in the aftermath of the events of September 11, 2001, with the creation of DHS, the implementation of Homeland Security Presidential Directive 5 (HSPD-5), "Management of Domestic Incidents," dated March 4, 2003, and the implementation of Presidential Policy Directive 8 (PPD-8) "National Preparedness," dated March 30, 2011. HSPD-5 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. PPD-8 directed the development of a national preparedness goal that identifies the core capabilities necessary for preparedness and a national preparedness system to guide activities that will enable the Nation to achieve the goal.

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, 2011, and 2013, 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 five National Planning Frameworks (i.e., prevention, protection, mitigation, response, and disaster recovery) provide guidance on Federal coordinating structures and processes to prevent, 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 HSPD-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 onscene 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 in 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 multiagency 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 the NRC's 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 the U.S. Department of Energy (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 multievent 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, the physical security measures implemented at NRC-licensed nuclear facilities provided for the protection of public health and safety against the design-basis threat for radiological sabotage as described in 10 CFR 73.1, "Purpose and Scope." Following the events of September 11, 2001, the nuclear industry significantly enhanced its defensive capability through voluntary actions by the licensees in response to NRC advisories and additional actions required by 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 codified its revised design-basis threat regulations on March 19, 2007, and updated the power reactor security regulations on March 27, 2009. These updated regulations incorporated provisions of the security orders and lessons learned during the implementation of the orders.

The NRC receives security-related information from the national intelligence community, law enforcement, and licensees, and it continually evaluates this information to assess threats to regulated facilities or activities. The NRC works with 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 Communications with Neighboring States and International Arrangements

The NRC has agreements with the United States' geographical neighbors, Canada and Mexico. The NRC's bilateral arrangements with nonneighboring countries also address and promote sharing of information on emergency preparedness and resources.

Since 2001, the United States has 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 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. 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 frequently regarding the emergency event.

Under its bilateral 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. The arrangement with Canada, the "Arrangement between the United States of America Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters," was most recently renewed in 2012 for a period of 5 years. The arrangement with Mexico, the "Arrangement between the Nuclear Regulatory Commission of the United States of America and the Comision Nacional de Seguridad Nuclear y Salvaguardias of the United Mexican States for the Exchange of Technical Information and Cooperation in Nuclear Safety and Research Matters," was most recently renewed in 2012 for a period of 5 years.

Because 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 have conducted several technical bilateral meetings in 2013, 2014, and 2015. Most recently, senior staff and managers from the Canadian regulator and the Mexican regulator observed a full Federal emergency exercise at the H.B. Robinson Nuclear Power Plant in South Carolina as a part of the U.S. Government's *Southern Exposure 2015* multiple-day exercise. During the exercise, the NRC hosted 52 foreign regulators, government officials, and nuclear utility staff from 13 countries, as well as observers from the IAEA and the Nuclear Energy Agency.

The NRC also routinely communicates with the IAEA and its Canadian and Mexican counterparts during its emergency drills. In addition, the NRC regularly participates in IAEA emergency preparedness and response conferences, technical meetings and consultancies in Vienna, Austria. The NRC also hosts several bilateral exchanges every year regarding emergency preparedness and response activities and emergency exercise observation with foreign regulatory bodies at the NRC Headquarters in Rockville, MD, and at U.S. nuclear power plants around the country.

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

After the Fukushima event, the NRC undertook actions to enhance emergency preparedness for licensees with respect to communications and staffing given a multiunit event and a prolonged SBO. The accident highlighted the need for licensees to identify the staff needed to respond to a multiunit 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 on assessing staff needs and communications to effectively respond to a multiunit event. This recommendation was identified as an activity that should begin without unnecessary delay (i.e., Tier 1).

The addressees were requested to assess both staffing and communications. Licensees were asked to evaluate 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 in 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
- 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
- their current staffing levels and the appropriate staff and positions to respond to a multiunit event given a beyond-design-basis natural event and to determine if enhancements were needed
- 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

All licensees submitted their communications assessments by October 31, 2012. NRC staff issued safety assessments documenting the staff's review to each licensee by July 2013.

Phase 1 of the staffing assessments had licensees evaluate their ability to respond to a multiunit SBO event utilizing existing processes and procedures. 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. The licensee responses to the RFI for the Phase 1 staffing assessments were received and evaluated by the NRC staff. The NRC staff issued acknowledgement letters to all licensees with multiunit sites.

Phase 2 of the staffing assessments has licensees assess staffing needs associated with implementation of mitigating strategies for beyond-design-basis external events. As such, the Phase 2 assessments have 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 on March 12, 2012. The majority of these assessments have been received. The remaining four assessments, with the exception of the FitzPatrick site that it is requesting relaxation until 2017, have been received by the agency. The NRC's review of these assessments will be completed by the end of 2016.

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 associated with the challenges faced in communicating with States and regional stakeholders. The staff has put considerable effort into improving its communication strategy with all response stakeholders and will continue to make enhancements when improvement opportunities are identified.

The NRC is focusing its efforts on the implementation of the Tier 2 NTTF Recommendation 9.3, which includes the following:

- adding guidance to licensees' emergency plans that describes the ability to perform a multiunit dose assessment (including releases from spent fuel pools) using the licensees' site-specific dose assessment software and approach

- conducting periodic training and exercises for multiunit and prolonged SBO scenarios

- ensuring that emergency preparedness equipment and facilities are sufficient for dealing with multiunit and prolonged SBO scenarios

The NRC staff determined that the mitigating strategies recommendation (NTTF Recommendation 4.2) addresses periodic training and exercises for multiunit 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 multiunit dose assessment using the licensee's site-specific dose assessment software and approach, and the NRC determined that all licensees had full capability to perform these dose assessments by May 2015.

In addition to the Tier 3 emergency preparedness items discussed in Sections 1.3.1 and 1.3.3 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 size.
- Evaluate whether potassium iodide should be prestaged beyond the current 10 mile zone.

In October 2015, the staff issued SECY-15-0137, “Proposed Plans for Resolving Open Fukushima Tier 2 and 3 Recommendations,” recommending that these two items be closed for the reasons stated below. In Staff Requirement Memorandum (SRM)-SECY-15-0137, dated February 8, 2016, the Commission approved closing these items.

The staff had conducted an extensive analysis of the emergency planning zone size in response to a petition for rulemaking numbered 50-104, “Petition for Rulemaking Requesting Amendments Regarding Emergency Planning Zone Size,” which can be found in www.regulations.gov (Docket ID: NRC-2012-0046). The NRC staff concluded that the current size of the emergency planning zones is appropriate for existing reactors (including multiunit sites) and proposed new reactors and that emergency plans will provide an adequate level of protection of the public health and safety in the event of an accident at a nuclear power plant. Furthermore, the staff noted that the current emergency planning zones provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the designated distances should events warrant such an expansion. In SRM-SECY-13-0135, “Denial of Petition for Rulemaking Requesting Amendments Regarding Emergency Planning Zone Size (PRM-50-104),” dated February 27, 2014, the Commission agreed with the staff’s proposal to deny the petition and directed the staff to publish the denial in the *Federal Register*.

Regarding the prestaging of potassium iodide beyond the current 10-mile zone, the staff plans to continue to monitor the studies being conducted by the World Health Organization, United Nations Scientific Committee on the Effects of Atomic Radiation, and the Fukushima Health Management Survey, and engage stakeholders as appropriate.

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 responsibilities of the U.S. NRC for siting, which include site safety, environmental protection, and emergency preparedness. This article discusses the regulations applying to site safety and their implementation, emphasizing regulations applying to seismic, geological, hydrological, meteorological, and radiological assessments. It explains environmental protection and reevaluation of site-related factors. It also addresses the Vienna Declaration on Nuclear Safety, which was issued in February 2015. Article 16 of this report discusses emergency preparedness and international arrangements, which would apply to Contracting Parties in obligation (iv) above. Finally, no changes to the current NRC practices associated with siting were identified as part of the NRC's Fukushima lessons learned initiatives.

The United States reviewed the results of the CNS 2015-2016 consultancy meetings that developed a template to support drafting Articles 17 and 18 of the contracting parties' National reports. The group of CNS experts helped correlate each subsection of Articles 17 and 18 with relevant IAEA safety requirements. The United States has taken into consideration the template and its supporting information. No changes to the U.S. National report were made as a result of this effort.

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.

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, "Factors To Be Considered When Evaluating Sites," 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, "Nonseismic 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 in 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, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," guides the staff in reviewing the site safety content of the applicant's safety analysis report. Review Standard (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 in December 2001; RG 1.91, "Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants," Revision 2, issued in April 2013; and RG 1.217, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," Revision 0, issued in 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 in 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 in March 2007, and RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, issued in 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 in 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 in August 1977. RG 1.59 is being revised and is expected to be issued for public comment in 2016.

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 in 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 in 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 in 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The seismic capacity of the certified designs is determined independent of any specific site but capable of being sited in most currently existing sites. Because a seismic probabilistic risk assessment requires site-specific hazards information, the NRC requires all new and advanced reactor designs to conduct a seismic margin analysis. This analysis evaluates the sequence-level ability of plant structures, systems, and components to withstand an earthquake with high confidence (i.e., 95 percent) of low probability (i.e., five percent) of failure capacities and fragilities for all sequences leading to core damage or containment failures. A design has an acceptably low level of seismic risk if the design-specific seismic capacity of the plant can withstand at least 1.67 times the ground motion acceleration of the design-basis safe shutdown earthquake.

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 current requirements for dose assessments from postulated accidents:

- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Revision 1, reissued in February 1983

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, issued in July 2000

- RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” issued in May 2003

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.

NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” issued in February 1995, provides updated information on light-water reactor accident source terms. In supplying 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.

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.

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 structures, systems, and components. 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, “Changes, Tests and Experiments,” 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 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, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” Revision 2, dated June 1974, and RG 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,” Revision 2, dated June 1974.

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 light-water reactor, the AP600 design. More recently, the agency has applied the practice to the AP1000 and Economic Simplified Boiling-Water Reactor designs with similar results and is applying it for all contemplated light-water reactor design certification application reviews, including the U.S. Evolutionary Power Reactor, the Mitsubishi Heavy Industries, Ltd.'s U.S. Advanced Pressurized-Water Reactor, and the APR1400. 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 calculations and reduced occupational exposures, providing the licensee regulatory margin. Since the issuance of 10 CFR 50.67 in 1999, the majority of 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 Environmental Protection Elements of Siting

This section explains the environmental protection elements of siting. It covers the governing documents and site approval process. Since the first operating plants in the United States received licenses, issues have arisen that must be considered in siting reviews for new facilities. 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, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," 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 in March 2000) guide the staff's environmental reviews for a range of applications, including site reviews for construction permits and operating licenses under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 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. On September 3, 2014, the NRC issued COL/ESP-ISG-026, “Interim Staff Guidance on Environmental Issues Associated with New Reactors,” to encompass the 2007 and 2010 updates. COL/ESP-ISG-026 will be incorporated into the next revision of the NUREG-1555. 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 nuclear power plant 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, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.” 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 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 Environmental Reviews

The NRC’s environmental standard review plan was first published in the 1970s. Since the 1970s, many changes to the regulatory environment have affected both 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 subsequent revisions to environmental standard review plans.

In the late 1980s, the NRC issued regulations for an alternative licensing framework to 10 CFR Part 50, which required a construction permit followed by an operating license. The framework in 10 CFR Part 52 introduced the concepts of approving nuclear power plant designs independent of sites and approving sites independent of these 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 has approved five early site permits and four combined license applications (for a total of seven licenses) 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 incorporates 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 developed an environmental impact statement that addresses the technical concerns raised by the Court and provides the National Environmental Policy Act analyses needed to support a revision to 10 CFR 51.23. In September 2014, the NRC issued a revised rule at 10 CFR 51.23 and its associated NUREG-2157, “Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel.” The revised rule adopts the generic impact determinations made in NUREG-2157 and codifies the NRC’s generic determinations regarding the environmental impacts of continued storage of spent nuclear fuel beyond a reactor’s operating license.

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 in 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 1978 decision on the Tennessee Valley Authority Yellow Creek Nuclear Plant, which determined that the authority of the NRC is limited in matters that are expressly assigned to U.S. Environmental Protection Agency (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 (Federal Water Pollution Control Act, Section 316(b))

- increased emphasis on greenhouse gases and climate change impacts

- EPA's Clean Power Plan rule, dated October 23, 2015 (80 FR 64966)

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, "Backfitting," 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 existing 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, 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 countries for siting new builds. The agency's current arrangements with its Canadian and Mexican regulatory counterparts for the exchange of information and experience serves as the mechanism for cooperative dialogue.

17.6 Vienna Declaration on Nuclear Safety

Consistent with the goals of the Vienna Declaration on Nuclear Safety, NRC regulatory requirements for siting have long reflected a defense-in-depth approach that requires all natural phenomena and manmade hazards at a potential site to be identified and properly accounted for in the siting, design and operation of the plant. This approach avoids siting of plants at problematic sites, and ensures that radiological doses from normal operation and postulated accidents will be acceptably low. As discussed in Section 17.2.1 of this report, siting regulations are implemented primarily through requirements in 10 CFR Part 100, Supbart B. In addition, the General Design Criteria in Appendix A to 10 CFR Part 50 further embody the defense-in-depth philosophy. General Design Criterion 2, “Design Bases for Protection against Natural Phenomena,” requires that the plant’s design basis reflects appropriate consideration of severe natural phenomena, including effects of normal and accident conditions. Integration of Criterion 2 with the other General Design Criteria provides assurance that the NRC’s approach to siting is consistent with the safety goals of the Vienna Declaration. This defense-in-depth approach is discussed further in Section 18.1 of this report.

The safety goals of the Vienna Declaration on Nuclear Safety are consistent with the NRC’s siting regulations. These regulations favor siting of nuclear power plants in areas of relatively low population density, with restricted-use zones around the plant that reflect the design characteristics of the plant (e.g., power level) and the atmospheric dispersion characteristics of the site. The plant’s design and operations must be protected from the effects of accidents at nearby civilian or military facilities, or from nearby transportation routes. Siting regulations also contain provisions to ensure that radiological doses from postulated accidents will be acceptably low. In addition, all natural phenomena that might affect the design or operation of the plant must be appropriately characterized, so that the plant’s design basis appropriately considers the most severe natural phenomena at the site, with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated. By taking this approach to protection against external hazards, the NRC’s regulations effectively discourage the siting of new plants at locations where there is an unacceptable risk of long-term offsite contamination or large releases requiring long-term protective actions.

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 existing 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.

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 U.S. 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. This section discusses requirements for reliable, stable, and easily manageable operation, specifically considering human factors and the man-machine interface, lessons learned from Fukushima, and addresses the Vienna Declaration on Nuclear Safety, which was issued in February 2015. Article 12 of this report also provides information on the human factors obligations.

Finally, the United States reviewed the results of the CNS 2015-2016 consultancy meetings that developed a template to support drafting Articles 17 and 18 of the contracting parties' National reports. The group of CNS experts helped correlate each subsection of Articles 17 and 18 with relevant IAEA safety requirements. The United States has taken into consideration the template and its supporting information. No changes to the U.S. National report were made as a result of this effort.

18.1 Defense-In-Depth Philosophy

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 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 verifies that the applicant’s quality assurance program is adequately implemented and that any pre-construction activities meet specified requirements in Appendix B to 10 CFR Part 50. Inspection Manual Chapter 2502, “Construction Inspection Program: Pre-Combined License (Pre-COL) Phase,” dated December 13, 2010, lists the 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, “Vendor Inspections,” dated October 3, 2013, 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. Inspection Manual Chapter 2503, “Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work,” dated July 5, 2012, describes these inspections. 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.

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 recent review of the Watts Bar Nuclear Plant, Unit 2, operating 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 Tennessee Valley Authority, is 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. Tennessee Valley Authority stopped construction at Watts Bar, Unit 2, in the mid-1980s; however, in 2007, Tennessee Valley Authority notified the NRC of its plans to resume construction.

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 ensures safety while preserving design and operational consistency between the units. However, considering the construction status of the unit, the NRC encouraged Tennessee Valley Authority 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 included safety design, environmental review, and inspection of construction activities.

Tennessee Valley Authority updated its initial 1970s operating license application. The NRC 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. Tennessee Valley Authority 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 May 2013. The NRC also 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.

To complete its licensing review in a timely and comprehensive manner, the NRC established dedicated teams at both its headquarters and regional offices for review and inspection of the Unit 2 activities. The staff independently reviewed Tennessee Valley Authority'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 in February 2009. The review identified the items that needed to be completed before issuance of an operating license. The staff published five additional supplements to NUREG-0847 documenting its review of the open items laid out in Supplement 21.

The NRC Region II office performed the necessary inspections and oversight activities for Watts Bar, Unit 2. It developed Inspection Manual Chapter 2517, "Watts Bar Unit 2 Construction Inspection Program," issued in February 2008, to provide guidance for these inspection activities. The NRC Region II office examined historical inspection records, employee concerns, operating experience, scope of new or rework, and construction deficiency reports. On January 30, 2015, the NRC issued IP 94302, "Status of Watts Bar Unit 2 Readiness for an Operating License." The objective of IP 94302 is to inform the Director of the Office of Nuclear Reactor Regulation of the following items: (1) completion of inspections necessary to support the findings required by 10 CFR 50.57(a)(1), 50.57(a)(2), and 50.57(a)(3)(ii), (2) any incomplete inspections or open items at the time that item (1) is communicated, and (3) any significant issues in the construction or testing that could affect the conclusions associated with item (1). The Region II Administrator provided notification of the above items to the Director of the Office of Nuclear Reactor Regulation on October 15, 2015.

Based on the findings of the NRC staff's review of the operating license application as documented in NUREG-0847 and its supplements, the NRC staff's inspection activities, the IP 94302 results, and the May 2013 final environmental impact statement, the operating license for Watts Bar Nuclear Plant, Unit 2, was issued on October 22, 2015, and is valid for 40 years. Initial criticallity at Watts Bar, Unit 2, was achieved on May 23, 2016.

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 in May 2002. The NRC used NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, issued in 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 in 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 AP1000 applications as well as the Vogtle, V.C. Summer, Fermi, and South Texas Project combined license applications. Reviews continue on the U.S. Advanced Pressurized-Water Reactor (US APWR) and the APR1400 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

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 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 possible failure mode 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 interim staff guidance documents for review of new and innovative digital instrumentation and control systems found in new reactor designs. The guidance also provided the industry with the expectations and criteria the staff uses to evaluate their designs and determine compliance with NRC regulations. The staff has been using this guidance, along with other existing guidance such as NUREG-0800, in its review of applications for design certifications and combined licenses. The staff has incorporated some of the interim staff guidance into formal NRC staff guidance in NUREG-0800 and associated RGs. All interim staff guidance documents on digital instrumentation and control can be found at http://www.nrc.gov/reading-rm/doc-collections/isg/digital-instrumentation-ctrl.html.

The staff has completed its safety reviews of the instrumentation and control systems for the AP1000, ESBWR, and Advanced Boiling-Water Reactor (ABWR) reactor designs as well as those for the Fermi, Unit 3, and South Texas Project, Units 3 and 4, combined licenses. The staff is in the process of reviewing the instrumentation and control design for the US APWR and APR1400 reactor designs and multiple combined license applications. The staff also has initiated the instrumentation and control ITAAC inspection activities for the AP1000 combined licenses, and preapplication activities on a small modular reactor design.

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. 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 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 US EPR and ABWR instrumentation and control technical expert subgroups, which are an international collaboration of regulatory agencies engaged in review of the US EPR and ABWR instrumentation and control designs. 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 in February 2002, and NRC Order EA-03-086, “Issuance of Order Requiring Compliance with Revised Design Basis Threat for Operating Power Reactors,” issued in 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, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants.”

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 operating power reactor 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. To meet the cyber security rule requirements, operating power reactor licensees had to submit a cyber security plan, including a proposed implementation schedule with interim milestones, to the NRC for review and approval by November 23, 2009, and combined license applicants are required to submit a plan in accordance with their overall license application. All operating nuclear power plant licensees met that submission deadline, and the NRC reviewed and approved all the plans. 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, the Institute of Electrical and Electronics Engineers, the National Institute of Standards and Technology, and the U.S. Department of Homeland Security.

In 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. The 5-year memorandum of understanding with the North American Electric Reliability Corporation was renewed in 2015. 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 share relevant operating experience and other related technical information. In 2010, the NRC entered into a 5-year 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. The 5-year memorandum of agreement was renewed in 2015.

The NRC has developed an oversight program for cybersecurity that includes an inspection program, inspector training, and a process for evaluating the significance of inspection findings. This was accomplished collaboratively with stakeholders, including members of industry and representatives from the U.S. Department of Homeland Security, the Federal Energy Regulatory Commission, and the National Institute of Standards and Technology. The NRC completed inspection activities related to the interim milestones in calendar year 2015. In 2016, the NRC, along with industry, is preparing for full implementation inspection activities that will begin in calendar year 2017.

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. In recent years, the ConE program was expanded to include reviews of events at operating reactors that were related to latent design and construction issues.

The staff's review of an operating experience or ConE event occasionally results in the issuance of INs or RIS. Recent examples include IN 2015-09, "Mechanical Dynamic Restraint (Snubber) Lubricant Degradation Not Identified Due to Insufficient Service Life Monitoring," dated September 24, 2015; IN 2015-04, "Fatigue in Branch Connection Welds," dated April 24, 2015; RIS 2015-08, and RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014.

The NRC staff values close cooperation with the international community for the exchange of information on design and construction of new reactors and continues to work closely with several countries that are currently building new nuclear power plants. As an example, the NRC ConE program staff is participating in a NEA Working Group on Regulation of New Reactors. The NRC staff also visits international sites under construction and participates in joint vendor inspections with regulators from other nations. The NRC values these partnerships and is committed to continuing its collaborative relationship with the international community.

The NRC also exchanges relevant construction related information with our international counterparts via the NEA's Construction Experience Program. Recent international events taken from the Construction Experience Program include issues associated with reactor vessel manufacturing anomalies, misinstallation of containment vertical tendon sheaths, and design mismatch of control room display windows of the plant monitoring and alarm system. In 2015, the NRC uploaded construction experience events to the Construction Experience Program database for issues related to the potential safety hazards caused by pipe support coating deviations and inadvertent damage to the V. C. Summer, Unit 2, AP1000 containment vessel.

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. As described in Sections 1.3.1 and 1.3.3 of this report, the NRC has issued requests for information for licensees to reevaluate their seismic and flooding hazards. Before the Fukushima accident, the NRC was evaluating two generic issues that involved natural hazards: Generic Issue (GI)-204, "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure," and GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimations in Central and Eastern United States on Existing Plants." Subsequently, the evaluation of these generic issues was incorporated into the Fukushima lessons learned reevalautions. As an additional safety enhancement against beyond-design-basis natural hazards, the NRC also issued an order requiring licensees to have mitigating strategies that preserve core cooling, SFP cooling, and containment. The staff's review of the Near Term Task Force 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, as discussed in Sections 1.3.1 and 1.3.3 of this report.

18.6 Vienna Declaration on Nuclear Safety

Consistent with the first principle of the Vienna Declaration on Nuclear Safety, new nuclear power plants licensed in the United States must meet safety, security, technical and financial qualification requirements in the NRC's regulations in 10 CFR Chapter I, including 10 CFR Parts 20; 50; 52; 30; 40; 70; 73; and 100; as well as 10 CFR Part 21, "Reporting of Defects and Noncompliance"; and 10 CFR Part 55, "Operators' Licenses." These NRC requirements govern the design, siting, construction, and operation of nuclear power plants. These requirements address the prevention and mitigation of accidents through the establishment of criteria for control and safety systems, such as the containment, reactor coolant systems, and emergency core cooling systems. Regulatory requirements exist to ensure adequate emergency planning to protect populations living within a 50-mile radius of nuclear power plants, and to evacuate populations living within a 10-mile radius of nuclear power plants in the unlikely event of a radioactive release. Each of these requirements were established with consideration of uncertainties to ensure that adverse consequences to the public are acceptably low. These regulations serve to prevent accidents and mitigate adverse consequences in a manner that effectively minimizes the potential for (and therefore addresses the risk of adverse consequences associated with) long-term offsite contamination.

In evaluating the design of a new reactor, the NRC assesses whether the prevention of accidents and mitigation of consequences provided for in the design meets the Commission's safety goals. In addition to addressing design basis events, the NRC has imposed a requirement on applicants seeking design certification under 10 CFR Part 52 to perform a PRA for their proposed design and requires them to provide a description and analysis of design features for the prevention and mitigation of severe accidents (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.) Combined license applicants are also required to address the design features for prevention and mitigation of severe accidents by considering site-specific conditions and factors, as well as a complete plant-specific PRA.

Because NRC requirements protect public health and safety through prevention of accidents and by mitigating releases in the event of an accident, including severe, beyond design-basis accidents, the risk of offsite contamination is rendered acceptably low.

The NRC uses deterministic and risk-informed requirements, as well as defense-in-depth principles, to achieve this goal. Defense-in-depth embraces a broad set of principles and requirements, including: (1) the need to prevent accidents from occurring and mitigating accidents if they occur (including robust emergency preparedness requirements), (2) the concept of multiple barriers against radioactive releases, (3) the application of the principles of independence, redundancy and diversity, which is implemented through requirements such as the "single failure" assumption, and (4) siting new nuclear power plants in lower population areas and areas with less adverse natural phenomenon characteristics. Section 18.1 of this report provides additional details regarding the NRC's defense-in-depth philosophy. Thus, the NRC's current regulatory approach provides reasonable assurance that there is a low likelihood of offsite contamination requiring long term protective measures and actions.

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 U.S. 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 the Vienna Declaration on Nuclear Safety, which was issued in February 2015.

Immediately after 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 are discussed in Sections 1.3.1 and 1.3.3 of this report.

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, "Domestic Licensing of Production and Utilization Facilities." This licensing process requires both a construction permit and an operating license. The additional licensing processes in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 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. To date, the NRC has issued five early site permits. According to this process, environmental and siting issues that have been resolved in the early site permit proceedings cannot be reopened during a combined license proceeding.

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 duration of a design certification is 15 years, and the certification may be renewed. The NRC has certified five standard plant designs under the design certification process: (1) General Electric’s ABWR, (2) Westinghouse Electric Company, LLC’s System 80+ (originally designed by Combustion Engineering), (3) Westinghouse’s AP600 design, (4) Westinghouse’s AP1000, and most recently, in October 2014, (5) General Electric-Hitachi’s ESBWR. In December 2011, the NRC staff issued amendments to the AP1000 and ABWR design certification rules. The NRC staff is currently performing the following two design certification reviews: (1) Korea Hydro and Nuclear Power’s APR1400 and (2) Mitsubishi’s U.S. APWR. In addition, the NRC staff has received two applications to renew the ABWR design certification. The NRC received one renewal application from General Electric Hitachi Nuclear Energy and a separate application from Toshiba Corporation. The NRC is actively reviewing the General Electric Hitachi renewal application. Toshiba requested withdrawal of its renewal application in June 2016.

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 must be met (collectively referred to as ITAAC) to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the applicable regulations. In 2012, the NRC issued its first combined licenses authorizing construction and operation of new nuclear power plants at two sites in the United States. The NRC issued combined licenses referencing the AP1000 certified design to Southern Nuclear Operating Company for two units at the Vogtle Nuclear Plant in Georgia and to South Carolina Electric and Gas Company and South Carolina Public Service Authority for two units at the V.C. Summer Nuclear Plant in South Carolina. In May 2015, the NRC issued a combined license referencing the ESBWR certified design to DTE Energy Company for one ESBWR unit at the Fermi 3 site in Michigan. In February 2016, NRC issued combined licenses to Nuclear Innovation North America for two ABWR units at the South Texas site in Texas.

After issuing a combined license, the NRC staff will verify that the licensee has performed the required ITAAC, 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 ITAAC. 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 that the public health and safety is adequately protected.

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, "Technical Specifications," 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, almost three-quarters 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 3, 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 in November 1980; NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," issued in January 1983; and NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," issued in 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 recommended 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 plants using the guidance in NUREG-0800, Section 13.5.2.1.

Furthermore, as discussed in Section 1.3.1 of this report, the NRC has ordered all power reactor licensees to develop mitigating strategies to respond to beyond-design-basis events at all units at a site for an indefinite period of time. A related rulemaking is underway that would codify that order, including the procedures and guidance for managing situations at multiunit sites, as well as incorporate a Fukushima lesson-learned initiative associated with strengthening and integrating emergency procedures. The proposed Mitigation of Beyond-Design-Basis Events rulemaking would require that licensees ensure smooth transition between each of their response guidelines, including FLEX support guidelines which are used for mitigating strategies, and emergency operating procedures. The implementation of the order is being inspected by the NRC after all units at a site come into compliance. The implementation of the rule requirements will be inspected at a later date, after the rule has been finalized.

In SRM-SECY-15-0065, "Proposed Rule: Mitigation of Beyond Design Basis Events," dated April 30, 2015, the Commission directed that SAMGs continue to be implemented voluntarily rather than being imposed as an NRC requirement. As such, each licensee has made a formal, written regulatory commitment to perform timely updates of the site-specific SAMGs with the vendor-specific owner's group technical guidance document and to integrate them with other emergency response guideline sets and symptom-based emergency operating procedures. Based on the Commission's direction, the NRC will provide periodic oversight of the SAMGs through the Reactor Oversight Process.

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. Section 50.120 of 10 CFR, "Training and Qualification of Nuclear Power Plant Personnel," requires licensees to establish, 18 months before fuel load, a variety of training programs for instrumentation and control, electrical maintenance and mechanical maintenance personnel, including engineering support personnel. The NRC verifies the adequacy of these programs during initial licensing. During the lifetime of the plant, availability of trained and competent engineers and technical support is revealed through equipment performance. The NRC's Reactor Oversight Process implements several IPs that focus on verifying the availability and operability of safety-related equipment and equipment important to safety. Inspectors may identify findings during these inspections. Licensees also report performance indicators, which are verified by the Reactor Oversight Process. 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, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 10 CFR 50.73, "Licensee Event Report System," 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 available on the NRC's public 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. Supplement 1 to NUREG-1022, Revision 3, published in September 2014, endorses NEI 13-01, "Reportable Action Levels for Loss of Emergency Preparedness Capabilities," dated July 2014. NEI 13-01 provides specific guidance for reporting under 10 CFR 50.72(b)(3)(xiii) and, as a result, reduces the need for engineering judgment.

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 in January 2002, and Information Notice 2009-27, "Revised International Nuclear and Radiological Event Scale User's Manual," issued in 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 in 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 through an internal Web site, which provides a centralized repository of links to databases relevant to operating experience on the NRC, 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. Events that may be of broader interest to technical staff may be summarized in an article for a periodic newsletter, or developed into standalone operating experience notes, which serve to provide a high-level summary of the issue and any actions being taken. 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 generic safety issue, or a revision of IPs.

The NRC's ConE program is described in Section 18.4 of this report. The NRC participates in the International Nuclear and Radiological Event Scale and the IAEA international reporting system for operating experience 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 rate 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 reactor 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 international reporting system for operating experience reports as they are posted. The NRC submits all U.S. relevant reactor-related generic communications to the IAEA international 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, "Standards for Protection against Radiation," 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[18] in 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," 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 Fifth National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management," Revision 4, issued in September 2014) is available on the DOE Environmental Management Web site.

In August 2013, the U.S. Court of Appeals for the District of Columbia Circuit ordered NRC to continue with the licensing process for DOE's Yucca Mountain construction authorization application, until Congress directs otherwise or there are no appropriated funds remaining. After the Court's decision, in January 2015, NRC completed the safety evaluation report for the application and found that DOE's application meets most, but not all, of the applicable NRC regulatory requirements. Specifically, requirements not met are related to certain conditions of land ownership and water rights. The NRC published a supplement to DOE's environmental impact statements in May 2016. "Supplement to the Department of Energy's Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada" evaluates the potential environmental impacts on groundwater and impacts associated with the discharge of any contaminated groundwater to the ground surface due to potential releases from a geologic repository for spent nuclear fuel and high-level radioactive waste at Yucca Mountain, Nye County, Nevada. The NRC's adjudicatory proceeding for the Yucca Mountain application, which must be completed before a licensing decision can be made, remains suspended.

19.9 Vienna Declaration on Nuclear Safety

The NRC relies on regulations in 10 CFR Chapter I and internally developed associated programs in granting the initial authorization to operate a nuclear installation and in monitoring its safe operation throughout its life. These regulations are generally consistent with IAEA and NEA safety standards and serve to prevent accidents and mitigate adverse consequences in a

18 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.

manner that effectively addresses long-term offsite contamination. Because NRC requirements protect public health and safety through prevention of accidents and by mitigating releases in the event of an accident, the risk of offsite contamination is rendered acceptably low.

Furthermore, a recent Integrated Regulatory Review Service (IRRS) mission conducted at the NRC found that the NRC has a number of processes in place, including a robust and mature inspection program, the analysis of the operating experience, the generic upgrades and regulatory changes, the use of risk informed regulation, and the license renewal rule, that meet the intent of a periodic safety review. The results of the IRRS mission and the alternate program that the United States employs in lieu of conducting periodic safety reviews are further discussed in Sections 8.1.5 and 14.1.5 of this report.

In conclusion, the NRC's regulatory practices are consistent with the principles of the Vienna Declaration on Nuclear Safety.

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

January 2016

INPO®

1. Executive Summary

The U.S. nuclear power industry established the Institute of Nuclear Power Operations (INPO or “the Institute”) in 1979 after the event at Three Mile Island Nuclear Station to promote the highest levels of safety and reliability (i.e., to promote excellence) in plant operation. INPO is a nongovernmental corporation that operates on a not-for-profit basis. Under United States tax law, the company is classified as a charitable organization that “relieves the burden of government.”

Since its inception, all utility 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 23 members. In addition, many utility organizations that jointly own these nuclear power plants are associate members. A number of international utility organizations and major suppliers also voluntarily participate in the Institute’s activities and programs.

In forming INPO, the nuclear power 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 past 35 years. For example, in the early 1980s the typical nuclear plant had a capacity factor of 63 percent, had experienced six automatic scrams a year, had high collective radiation dose, and had experienced numerous industrial safety accidents among its staff. Today, the median industry capacity factor is above 92 percent, most plants have no automatic scrams a 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. This report includes an overview of the industry actions to address extreme external events and the Institute’s role in response to the accident at Fukushima.

As discussed in Part 1 of this report, the United States ratified the Convention on Nuclear Safety (CNS) in 1999 and has been actively participating in its peer review activities. The conclusions from the review of the 2013 U.S. National Report at the Sixth CNS review meeting in April 2014 were very positive. The United States was a member of Country

Group 1. Country Group 1 identified the following challenges regarding the United States nuclear industry strategy:

- Continuous improvement in self-awareness. In 2014, INPO established a continuous monitoring program with the purpose of maintaining an accurate picture of plant performance between evaluations. This program helps INPO identify early signs of performance decline so that actions can be taken to prevent further decline. Additional information can be found in Section 7.d of this report.

- Effective use of operating experience. INPO's analysis and information exchange programs help improve plant safety by identifying the causes of industry events that may be precursors to more serious events. INPO requires that stations share operating experiences and lessons learned with INPO. INPO staff then analyzes and communicates this information to the industry through a variety of methods and products. Additional information can be found in Section 7.c of this report.

- Maintaining proficiency of the nuclear workforce. 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. Section 7.b of this report describes the training and accreditation programs of the nuclear industry and INPO's role in this area.

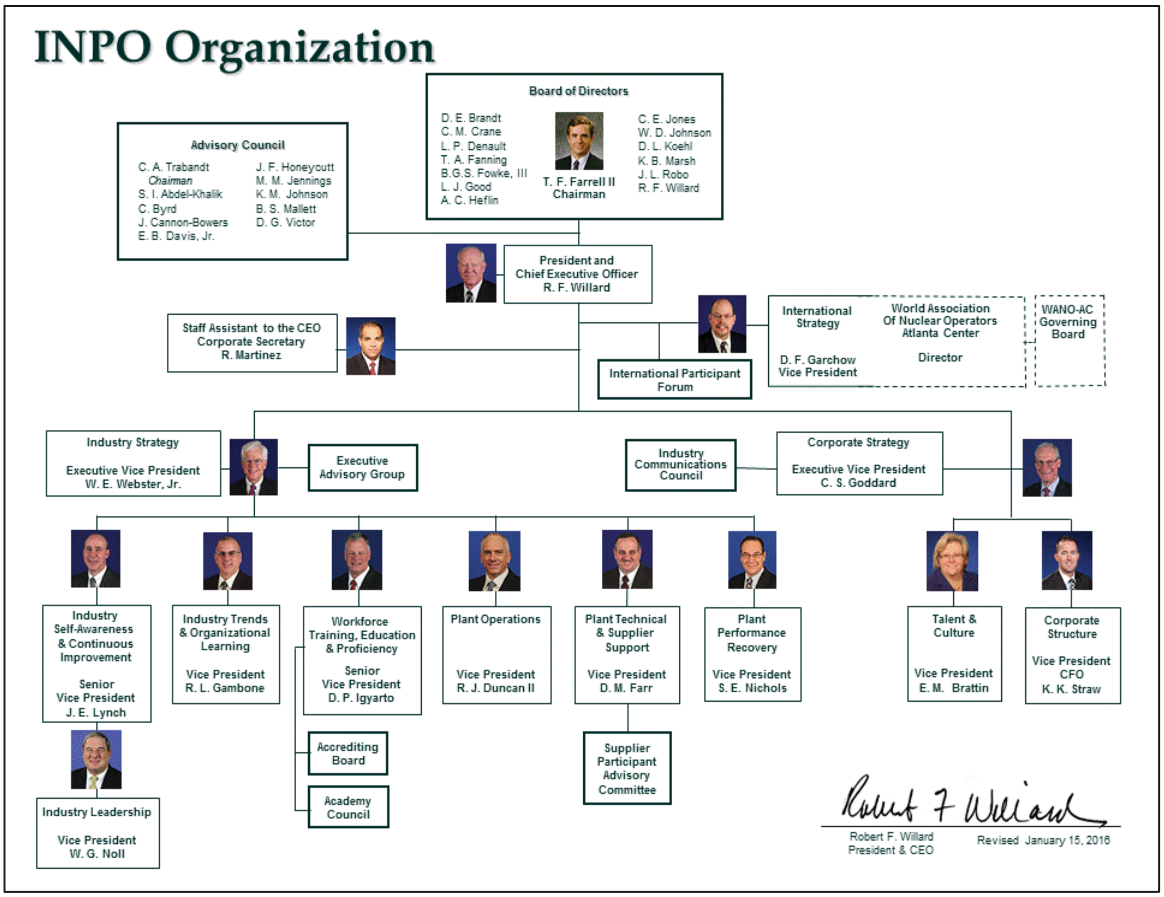
- Sharing best practices of supplier and nonnuclear support. INPO maintains a supplier participant program in which supplier participant members share best practices and operating experience. In October 2014, INPO issued an industry standard for nuclear suppliers, INPO 14-005, "Excellence in Nuclear Supplier Performance." This standard describes the essential principles and attributes that support achieving excellence in the services and products provided by nuclear suppliers. Supplier Participants completed their first self-assessment in August 2015 against INPO 14-005. Several work teams were created to provide recommendations to the supplier community on how to close identified gaps.

- Site resiliency against external events. INPO developed and communicated lessons learned from the Fukushima Dai-ichi accident in the form of INPO Event Reports (IERs). INPO conducted review visits at each domestic site to verify the implementation of the recommendations from the various Fukushima related IERs. INPO continues to evaluate site readiness against extreme external events during plant evaluations and peer reviews. Additional information can be found in Sections 7.a.iii and 9 of this report.

- Quickly and sustainably recovering lower performing plant. INPO maintains a special focus program for stations where behaviors and results are not representing high levels of performance or stations experiencing a steep decline in performance that increases the likelihood for a significant event. Section 7.e of this report describes this program.

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 senior executives from INPO's member organizations, provides overall direction for the Institute's operations and activities. Currently, the board comprises 13 chief executive officers (CEOs) 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 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. Members include prominent educators, scientists, engineers, business executives, and experts in organizational effectiveness, human relations, and finance.

The industry actively participates in the oversight of INPO's programs. Representatives from member utilities serve on the Executive Advisory Group, the Academy Council, and the Industry Communications Council. The Executive Advisory Group, which comprises 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 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.

Six core characteristics enable INPO's self-regulation model to be effective in fostering the highest standards of safety and reliability at U.S. nuclear power plants:

- CEO engagement: A fundamental element in founding INPO was the personal involvement and support of member CEOs. Today, that same level of support and involvement remains fundamental to INPO's continued impact on the industry.

- Nuclear safety: INPO's mission of promoting the highest levels of safety and reliability – to promote excellence – in the operation of commercial nuclear power plants has not wavered. Nuclear safety is at the forefront of every INPO activity. Additionally, the distinction between excellence and regulatory compliance is foundational to continuous improvement in nuclear safety and reliability.

- Broad industry support: The nuclear industry was involved in developing standards of excellence and is committed to meeting those standards. The industry accepts that as part of the self-regulation model, its nuclear stations are subject to onsite evaluations that involve participation by industry peers. The evaluations are intrusive, comprehensive, and performance-based. The industry also supports and participates in self-regulation through involvement with advisory groups, industry task forces and working groups, and by loaning employees to INPO. Through such involvement, participants gain firsthand experience and knowledge on improvement opportunities at their own sites and also increase their understanding of INPO's role and the importance of self-regulation.

- Accountability: INPO's formal process of evaluations and assessments provides a basis for continuous industry improvement that includes peer pressure and the identification and targeting of plants that require special assistance to help improve performance in key areas. Furthermore, in rare instances utility insurance rates can be impacted as a consequence of INPO evaluation results.

- Independence: Although INPO is part of the nuclear power industry, it remains independent. The Institute establishes high industry standards and distinguishes clearly between its evaluative role and other collaborative interactions and activities with its members.

- Confidentiality: INPO and its member utilities recognize that for continued success, it is essential that the nuclear industry maintain a healthy environment for peer review and self-improvement. Candid interactions with utility staff, which are central to the evaluation process, are predicated on the assurance the information will be used privately and constructively. Misuse of information contained in INPO reports by individuals outside the utility would have a detrimental effect on INPO's ability to obtain information and to identify needed improvements.

The Institute is committed to a long-term strategic design that outlines the ways and means by which it will fulfill its mission through 2023. The strategic design takes into account the current state, the desired end state, and potential barriers in shaping desired outcomes in three separate but interrelated areas: INPO's corporate responsibilities, U.S. nuclear industry performance, and international nuclear industry performance. Defined within its strategic design are priorities and measurable outcomes that guide the application of INPO's limited resources.

<u>INPO's Corporate Strategy</u>

INPO is guided in its corporate responsibilities by the strategic bases for shaping U.S. and international industry performance, together with traditional corporate tasks of developing its workforce:

- INPO is committed to attracting top-performing employees whose talents match Institute and industry needs.

- Understanding that a strong culture has a powerful influence on behaviors and performance, the Institute strives to instill a culture that emphasizes integrity, accountability, and high performance. It also ensures employees are equipped with the necessary sensitivities and flexibility to navigate cultural differences encountered both domestically and internationally.

- Clear, well-executed processes guide INPO's application of resources. The Institute ensures that processes support its mission and provide reliable and predictable results.

- INPO employs a matrixed organizational structure whereby its staff supports cross-functional initiatives. This requires an internal work environment that is stable, complete, unambiguous, and consistent across the organization, while maintaining the flexibility and scalability to adapt to changing needs.

<u>U.S. Industry Strategy</u>

In pursuit of nuclear safety, reliability and operational excellence, INPO sets performance standards for the industry. It then measures industry performance against those standards and facilitates performance improvement through education and training, widespread sharing of best practices, lessons learned, and assistance. Finally, when it must, INPO exercises the self-regulatory authority granted by its member utilities.

INPO's industry-facing strategy currently addresses six challenge areas:

- Fundamental attributes of high-performing industries include self-awareness and the capability to continuously improve. Industry management must be proactive, intrusive, and knowledgeable to reduce recurring or long-duration shutdowns, as well as recognize the presence of key risk factors that can lead to significant events. It is vital that a high level of awareness is maintained regarding worker proficiency and that training be applied to mitigate proficiency shortfalls and to minimize human error. The more that leaders are educated, trained, developed and committed to knowing their plants and adapting to inevitable variances in performance, the less susceptible the nuclear industry will be to unanticipated, negative outcomes.

- An operating experience culture is paramount to ensuring that the nuclear industry remains alert to adverse safety and reliability trends. In embracing lessons learned, operating experience must become pervasive and central to management and worker decisionmaking. Achieving long-term performance goals requires that management recognizes the merits of operating experience and transfers its lessons down to the worker level.

- Considering the vital importance of a knowledgeable workforce to nuclear safety, training must be of the highest quality to ensure industry needs are met. This requires an integrated approach to sourcing, educating, training and qualifying workers. A broad array of management, leadership, and training approaches is necessary to help sustain worker proficiency and minimize human error. Leaders must prepare the workforce to adapt to changing conditions, including changes in site performance, to ensure the right management and leadership mix, along with the right qualifications, are in the right place at the right time.

- Suppliers and nonnuclear support organizations provide vital functions for the nuclear industry and can directly impact overall safety and reliability. Therefore, it is vital that INPO encourages the management of such groups to improve these areas — to achieve uniformity and quality for the nuclear industry.

- Lessons learned from the Fukushima Dai-ichi accident establish that nuclear sites need to be resilient against extreme external events. By applying concepts of defense-in-depth, sound response strategies, planned outside assistance, and having committed leadership, the nuclear industry will better withstand such events.

- Corporate leadership and site leadership are pivotal in plant recovery, workforce alignment, and sustainability. They must have an unwavering focus on finding and fixing problems by being intrusive, engaged and adapted to a site’s culture while managing challenges and distractions. Management teams will be better equipped to lead recoveries by developing the applicable leadership skills in advance of such circumstances and by preidentifying the means to augment leadership capability or capacity shortfalls.

<u>International Industry Strategy</u>

Internationally, INPO leverages the World Association of Nuclear Operators (WANO) for its widespread global reach, facilitates like-minded international regulatory and self-regulatory organizations that promote safe nuclear operations, and focuses on international partnerships for which the benefits to industry safety are most impactful.

- Leveraging WANO's global reach, INPO liberally shares its informational products throughout the international industry. As a WANO member, INPO participates in peer reviews, technical assist visits, and other WANO activities.

- INPO associates with and facilitates improvement of like-minded organizations, such as other national-level self-regulators, the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA), so that synergies in operational safety approaches may be realized.

- To the maximum extent, INPO will apply its limited resources directly to international utilities deemed to pose the highest safety risk, as well as learn from the most proficient ones.

Financial and Human Resources

The 2015 operating budget for INPO of $116 million 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 350 is augmented extensively by industry professionals who serve as loaned employees or international liaison engineers on assignments of 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 advisers. 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 2015, the industry provided INPO with more than 970 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 could 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, after 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 effectively bring 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 it would have to work closely with the NRC while not becoming or appearing to become an extension of, or an adviser 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 covering 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 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 reflecting 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 the following information on programs and activities at the Institute:

- copies of selected generic documents
- access to other pertinent information, such as the INPO Consolidated Event System (ICES) 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. The Institute'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 other industry standards of excellence. 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 does not respond 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 utility'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, the 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 follow up 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 externally 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 (to gain commercial advantage, for example). 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 the Institute 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 electric deregulation, the industry has reaffirmed INPO's mission to promote the highest levels of safety and reliability in the operation of nuclear power plants. Even with U.S. utilities now in competition in certain areas, these plant operators 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 the sharing of industry information by including participation of industry peers in several of the Institute's programs — plant evaluations, training and accreditation, analysis and information exchange, and plant recovery. INPO communicates and shares information through a variety of methods, including the secure Nuclear Network® member Web site, 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.***

In 2012, INPO distributed a report titled "Traits of a Healthy Nuclear Safety Culture." This document was developed through a collaborative effort of the U. S. and international nuclear operating communities, and representatives from NRC, the public, and INPO staff. The report replaced the INPO report "Principles for a Strong Nuclear Safety Culture," issued in November 2004.

In April 2013, two addenda were developed and distributed in support of the Nuclear Safety Culture Traits. Addendum I is titled "Behaviors and Actions that Support a Healthy Nuclear Safety Culture." It includes the behaviors and examples found in the "Traits of a Healthy Nuclear Safety Culture," but sorted by organizational level and attribute. Addendum II is titled "Cross-References for Traits of a Healthy Nuclear Safety Culture." It cross-references the Traits to the INPO principles document, NRC safety culture components, and IAEA safety culture characteristics.

INPO activities reinforce the primary obligation of the operating organization's 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 boardroom 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 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 problems at other plants. The 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:

- discussing a case study on the event with all managers and supervisors in the nuclear organization

- periodically conducting a self-assessment to determine the organizational respect for nuclear safety

- identifying and resolving abnormal plant conditions or indications that cannot be readily explained

Safety culture is thoroughly examined during each plant evaluation. INPO expects each evaluation team to review the 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. Operations, Activities and Actions

In the execution of its strategic design, INPO conducts a broad spectrum of large-scale operations, such as plant evaluations and training accreditation visits, recurring activities (for example, processes), and one-time actions. Several of these are long-standing, cornerstone INPO efforts, including those described below.

a. Evaluation Programs

Members host regular INPO evaluations of their nuclear plants approximately every 2 years. The INPO evaluation teams periodically conduct additional 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 basis 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 2015, U.S. utilities were visited for 31 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
- emergency preparedness
- fire protection
- industrial safety

The teams also evaluate cross-functional performance areas (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
- work management
- performance improvement (learning organization)
- organizational effectiveness

Teams also evaluate the following foundational areas:

- leadership
- nuclear professionalism

As part of the process, an evaluation team looks at important aspects of a site's quality assurance and oversight programs to ensure that these programs provide confidence that the plant is satisfying the requirements for activities important to nuclear safety.

Team leaders provide a focal point for the evaluation of station management and leadership by concentrating on evaluating leadership, teamwork, organizational effectiveness, safety culture, technical conscience, 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 includes, where practicable, observations of refueling outages, plant startups, shutdowns, major planned evolutions and planned fire and emergency preparedness drills.

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 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 station's performance compares 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. Several conclusions can also be drawn from evaluations over the years. Good performing plants (Category 1 and 2) show strong leadership, are self-critical, do not tolerate complacency, are operationally focused, have exceptional equipment performance, and effectively use training to improve performance. Category 3 and Category 4 stations may include leaders who do not set high standards, possess 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.

INPO technical department managers also provide an area performance summary in which they provide perspective on current performance in their area as compared with the industry. Each summary includes an articulation of the trend and the trajectory of performance.

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 in place of an INPO plant evaluation. They use a methodology and performance objectives 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 shut down 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 nuclear stations request that INPO conduct corporate evaluations at 5 to 6-year intervals. The evaluations reflect the important role of the corporate office, as well as corporate nuclear and nonnuclear leaders, in supporting safe and reliable nuclear operation. INPO conducted six corporate evaluations in 2015.

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 include the following:

- organizational effectiveness, including leader and team behaviors, as well as the effectiveness of programs, processes and the implementation of the management model

- 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 use these briefings as an input to their assessment of operational, project and enterprise 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. INPO conducted 94 review visits in 2015.

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 requiring 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.

PWR Materials Review Visits

INPO initiated review visits targeting the steam generator in 1996. In the early 1980s, steam generator tube leaks and ruptures contributed 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. The industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), issued 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.

Subsequently, in 2003, a primary systems integrity review visit was launched in response to a number of notable events associated with leakage from PWR borated systems resulting in additional oversight by the NRC and INPO. In some cases, these leakage events 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.

Industry performance has steadily improved in both steam generators and primary system integrity as evidenced by the lack of safety-significant events and events that contribute to lost generation. Utility programs addressing these areas are mature.

In 2012, the two programs were combined to form the PWR materials review visit to capture all aspects of the industry initiative codified in NEI 03-08, “Guideline for the Management of Materials Issues.” This initiative encompasses the Steam Generator Review Visit Program, the Materials Reliability Program, and other programs directly dealing with primary system materials. While the review visit scope and team size is larger, the objective remains the same: ensure nuclear safety and plant reliability are not compromised because of weakness associated with the primary pressure boundary, including the steam generators. However, the focus on establishing effective station programs and capturing newly implemented industry guidance has been replaced with an emphasis on program implementation, capturing ongoing industry operating experience, and performing forward-looking trending to ensure material degradation is proactively managed.

In 2016, the scope of the PWR materials review visit is being expanded to take an even broader look at materials degradation and will include flow accelerated corrosion programs and buried pipe and tank integrity.

BWR Materials 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.

Overall industry performance improved as evidenced by the lack of safety-significant events and events that contribute to lost generation.

In 2016, the scope of the BWR vessel and internals review visit is being expanded to take an even broader look at materials degradation and will include flow accelerated corrosion programs and buried pipe and tank integrity. In conjunction with this scope change, the name of the review visit is being changed to reflect the broader scope to BWR materials review visit.

AC Power Source Reliability Review Visits

In 2014, INPO combined the transformer, switchyard and grid review visit program with the emergency diesel generator review visit program to support the industry focus area of AC power reliability. There are four to six loss of offsite power (LOOP) matrix reviews targeted per year prioritized on a performance basis. These reviews, termed AC power reliability review visits, integrate the scope of the transformer, switchyard and grid review visit program and the emergency diesel generator visit programs with additional focus on program and procedures relied on to prevent, detect, and mitigate LOOP and station blackout events. Team peer selection will include individuals with transmission system and emergency diesel expertise.

To ensure consistent monitoring of performance, AC power reliability will remain an industry focus area on evaluation teams through review of plant events. In addition, a new indicator was developed to reflect AC power reliability for the industry and individual sites. The metric combines LOOP events and emergency diesel generator performance and availability on a 2-year rolling average.

INPO is also actively partnered with the North American Transmission Forum (NATF) to develop common expectations and risk assessment tools for the switchyard and grid system interface. In 2014, INPO, NATF and EPRI began joint efforts focused on AC power reliability. A May 2015 industry summit meeting highlighted actions each work group is taking to mitigate challenges to AC power reliability. As a part of the AC power source reliability strategy, INPO is also engaged with EPRI in the industry

Flexible Power Operations initiative for plants requested to accommodate renewable resource power contribution to grid load demand.

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, extended an outage, or both, had doubled from 1999 to 2003. During this time, unplanned scrams caused by generator problems increased to around five a year from the previous average of two a year. These review visits were suspended once industry performance improved and resources were shifted to emergent industry issues.

In 2016, INPO will resume monitoring main generator performance based on an increase in challenges to reliability of generator excitation and stator water cooling systems. Initially, main generator health will be reviewed on plant evaluations. Teams will focus on performance and condition monitoring to ensure that the generator is operating within design parameters and that monitoring is in place to detect early signs of equipment degradation.

Emergency Preparedness

Between 2007 and 2015, INPO conducted emergency preparedness review visits at U.S. nuclear stations and provided recommendations for improving readiness to respond to radiological and other site emergencies. Actions were also taken to improve timeliness and accuracy of event classifications, notifications, and protective action recommendations; strengthen drill programs; and increases in emergency response organization staff training.

In 2012, INPO established a new division focusing on emergency response. The division is responsible for continuing improvements in emergency preparedness, as well as working in concert with the U.S. industry during implementation of post-Fukushima changes to increase resilience for beyond-design-basis external event threats. Since 2013, INPO conducts review visits at each domestic site verifying implementation of recommendations from the various IER related to the Fukushima accident.

In 2015, fire protection was added to the division’s scope, and emergency response evaluators began examining emergency preparedness and fire protection during WANO peer reviews. Starting in 2016, emergency management and fire protection will form a single department under the plant operations division. Emergency management and fire protection evaluators will review these areas as part of evaluation teams.

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 National Academy for Nuclear Training (NANT or "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 achieve 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 6 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 self-evaluations by members with assistance from INPO staff, onsite evaluations by teams of INPO and industry personnel, and decisions by the independent National Nuclear Accrediting Board. Members 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 2015, the industry updated the accreditation objectives to better focus on the fundamental aspects of accredited training programs. Also, a training evaluator was added to the plant evaluation process to provide a more distinct look at the linkage between knowledge, skill, and performance. Together, these changes are designed to provide a clearer picture of the health of station training programs and the impact on worker proficiency.

The 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 and reliability in the nuclear industry. 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 a report titled “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.

The 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 February 2006, INPO launched the National Academy for Nuclear Training e-Learning (NANTeL) system. Using Web-based technologies that allow distance learning, NANTeL system training includes a variety of courses and proctored examinations. These include courses for plant access, radiation work, and industrial safety, maintenance, and engineering qualifications. Over 8 million courses have been completed in NANTeL since its inception.

In 2015, the Leadership and Team Effectiveness attributes, INPO 15-005, were integrated into these courses and seminars. Examples of courses and seminars conducted are as follows:

- 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
- nuclear operational risk course for managers
- operations supervisor professional development seminar
- first-line leadership seminar
- next-level leadership seminar
- seminars for new managers

INPO continues to work with the industry to develop and deliver training to address industry needs. For example, INPO recently developed an instructor training and certification program and a regulatory exam authors course for the industry. In 2016, an industrywide simulator instructor certification program will also be available.

c. Analysis and Information Exchange Programs

The analysis and information exchange programs help 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 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 Operating Experience 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, tagged, and analyzed for significance; those with generic applicability are disseminated to the industry in one or more of the following forms:

- Level 1 IER
- Level 2 IER
- Level 3 IER
- Level 4 IER

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 via the ICES. 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. 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 Level 1 and Level 2 IERs. During onsite plant evaluations, INPO teams follow up on the effectiveness of each station's actions in response to the recommendations. Topics of Level 1 and Level 2 IERs in recent years include integrated risks to plant viability, weakness in reactivity control, recurring electrical shock events, and ineffective dose monitoring.

Members should review and take actions, as appropriate, on IERs. 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 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" 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. 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 Performance Objectives and Criteria 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, "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. There are also principles documents providing expectations for excellence in areas including leadership and team effectiveness, integrated risk management, effective technical conscience and nuclear supplier performance.

- 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.

- 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. 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.

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 2015, more than 2,100 industry personnel participated in more than 70 seminars and workshops at INPO.

iv. Nuclear Network® System

Nuclear Network is an international electronic information exchange for sharing nuclear plant information. It is a major communication link for the operating experience program and WANO event reporting system. The system transmits operating experience information 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.

v. 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. Plant data are then consolidated for trending and analysis purposes. Industry wide 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.

vi. Equipment Performance Data

The industry reports equipment performance information to ICES, and 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 industry wide performance problems. The Institute also makes the data available to the NRC to support equipment performance reviews by the regulator.

vii. 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.

viii. 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. 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 reports include event classification and notification, mechanical maintenance performance, operator performance proficiency, and piping and tubing leaks. 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.

d. Comprehensive Performance Monitoring Program

In the second half of 2014, INPO established a performance monitoring program that uses all available data and information in combination with targeted, systematic assistance visits to develop an ongoing, comprehensive picture of plant performance between evaluations, such that timely and effective action can be taken to avoid declines. Preventing declines is part of an overall strategy help the industry achieve a condition that all stations operate at a high level of performance, meeting industry goals, with no significant events, or long-duration shutdowns, and no training program accreditation probations.

A team of performance monitoring leaders (PMLs) and support personnel continuously review and analyze performance data of stations to identify subtle signs of decline. Additionally, a core team of assigned INPO subject matter experts continuously review and analyze performance data pertaining to their specific functional areas. Performance is reviewed by all PMLs and support team members twice a quarter. The INPO senior leadership team reviews and challenges on a quarterly basis the picture of performance presented by the PML. Each PML is responsible for monitoring approximately eight stations that are grouped by fleet organizations. When signs of decline are identified, the PML works with station leaders to craft an assistance plan to arrest the decline and improve performance.

The methodology to achieve the comprehensive monitoring objective has three dimensions:

- Monitoring: Monitoring leaders use all available data and information to characterize station performance. Integrating data with plant observations and insights from other touch points allow the PML to develop an integrated picture of station performance. Credible trigger points are used to identify early gaps that require intrusion.

- Engage: Monitoring leaders engage station leaders, primarily site vice presidents, to understand the station leader's awareness of performance issues and the effectiveness of corrective actions. Station leaders receive an INPO Performance Summary Report (IPSR) twice a quarter. This report summarizes the current integrated picture of station performance from INPO's perspective. Additionally, the chief nuclear officers and chief executive officers of each utility receive a quarterly performance summary letter that provides a high-level paragraph describing the current performance and trajectory of their stations.

- Intervention: Intervention is required to shape performance improvement using a graded and specific approach. There are two levels of intervention – elevation for narrow shallow gaps, and escalation for wider deeper, or cross functional gaps. Targeted elevation or escalation plans are developed with the station leadership team to focus industry and INPO efforts to turn performance. In the case of a precipitous decline, the plant may be assigned to the plant performance recovery organization. Performance recovery uses a different process that relies more on direct observations of station performance and more interactions with station leaders.

e. Special Focus Program

A special focus program is in place for stations where behaviors and results are not representing high levels of performance or stations experiencing a steep decline in performance that increases the likelihood for a significant event. In most cases, improvements are needed in a number of areas and significant weaknesses may exist. A recovery plan is developed as part of an integrated strategy to improve plant performance. Routine and frequent interactions occur with station and utility/fleet leaders. Specific INPO and industry assistance, including structured interactions with the site, utility executives, the utility CEO and utility board of directors are required to address persistent shortfalls or a significant decline in performance.

Between evaluations, a station can request and receive assistance in specific problem areas to help improve plant performance. Assistance resources are provided using a graded approach that provides a higher priority to those plants that need greater performance improvement. This assistance is targeted for specific technical concerns and for broader management and organizational issues. Although a station generally requests assistance, INPO may, in some cases, suggest assistance in a specific area to stimulate improvements.

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 2015, INPO provided more than 270 assistance visits using over 225 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 2015 involved supplier participants, with a focus on supplemental personnel and fuel performance. In addition to assistance visits to stations for specific functional areas, INPO Performance monitoring representatives monitor and trend station performance using data provided by the stations on a quarterly frequency. Over 120 data elements involving a wide variety of operating, maintenance and other stations activities are reviewed for possible adverse trends, and INPO monitoring representatives interact with station management for early signs of performance decline. INPO teams also made multiple assistance visits at stations designated as special focus.

Effectiveness reviews performed by INPO approximately 6 months after assistance visits show that the visits are highly valued by station management and contribute to improved performance.

f. New Plant Deployment

No new nuclear plants have been built in the United States for many years. 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, some U.S. utilities are once again planning or have begun new plant construction. To support this effort, INPO formed a new plant deployment group several years ago and began to engage with the nuclear industry and plan for the Institute's involvement though application of its cornerstone programs.

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 have used this document in their review of the new designs.

INPO also engaged utilities in a series of benchmarking trips to international utilities and plant designers in several countries as well as unrelated industries such as an aircraft company, a coal plant with advanced control systems, and a company that uses modular technology to build refinery installations. 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 has issued several reports to its members that features the information gathered from these trips.

In an effort to further support utilities with construction projects underway, INPO issued a nuclear exchange on operational readiness lessons learned from a recent plant startup. With these lessons in mind, INPO issued an operational readiness guidance document and uses this guidance to conduct periodic review visits at units preparing for initial operation, which allows for the early identification of potential gaps and help to ensure the success of the project.

8. 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 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 Atlanta'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-Atlanta Center teams of U.S. and international peers conduct reviews at the request of INPO members to identify strengths and areas for improvement associated with nuclear safety and reliability. A WANO-Atlanta Center peer review conducted at a U.S.-INPO member plant is performed in place of an INPO plant evaluation.

- U.S. nuclear utilities share their operating experience with INPO, and WANO-Atlanta Center and passes along the U.S. operating experience to WANO. The operating experience sharing provides detailed descriptions of events and lessons learned to member utilities worldwide. International operating experience comes to WANO and in turn is entered into ICES to share with the U.S. members.

- WANO-Atlanta Center collects, trends, and disseminates performance indicator data to facilitate goal setting and performance trending and to encourage emulation of the best industry performance.

- WANO-Atlanta Center conducts technical support missions to allow direct sharing of plant operating experience and ideas for improvement.

- WANO-Atlanta Center, with the support of INPO, designs professional and technical development courses, seminars, and workshops to enhance staff development and to share operating experience.

The U.S. nuclear power 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.

9. 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 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 helped ensure that lessons learned were identified and well understood, and that response actions were effectively coordinated and implemented throughout the industry.

Separately, the NRC conducted an independent assessment and has initiated actions to ensure that its regulations reflect lessons learned from the Fukushima events.

The primary objective of the industry response is to maintain and improve already high levels of operational safety and reliability, while applying the lessons from the Fukushima Dai-ichi nuclear accident to strengthen resilience for external events. 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.

- 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.

In addition to directly supporting the industry response strategy to the Fukushima accident, INPO issued several INPO event reports (IERs) providing recommendations for addressing lessons learned from Fukushima. In general, the recommendations were crafted to be compatible with and supportive of actions required by the NRC. The IERs are summarized below:

(1) IER 11-1, "Fukushima Daiichi 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 were caused by factors directly affecting nuclear safety that were outside the design basis for the facility. Immediate actions by the U.S. industry were 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:

- 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 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.

• Verify that the capability to mitigate SBO conditions required by station design is functional and valid, as follows:

  - 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.

• 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 and that accessible doors, barriers, and penetration seals are functional.

• 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 Daiichi Nuclear Station Spent Fuel Pool Loss of Cooling and Makeup," dated April 25, 2011 and revised October 28, 2014

The earthquake and tsunami caused the loss of all station power supplies. 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 following 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:

- 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.

- 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 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.

- 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.

- Verify the adequacy of station abnormal operating procedures for responding to the loss of SFP cooling or inventory.

  - Ensure these procedures include actions to monitor SFP level and contingencies to monitor SFP temperature and area radiation readings when necessary because pool cooling or level cannot be maintained.

  - Provide the capability to make up inventory to the SFPs during a loss of all AC power.

  - Verify that the guidance in the abnormal operating procedures can be implemented during severe weather, seismic events, loss of control room, and flood conditions.

- Revise station emergency operating procedures or other event-based procedures to include a precautionary statement that SFP level and temperature should be monitored.

- If dry casks are used for storage of spent fuel, establish procedures to verify the cask condition following severe weather, seismic events or flooding. These procedures should include visual inspections to identify cask damage that could result in a loss of containment, shielding or cooling functions. Procedures should also include area radiation surveys to identify any deviation from normal background levels and should identify response actions if abnormal conditions are found. Include use of these procedures in training for applicable personnel.

- Establish a periodic surveillance test or maintenance activity to verify the functionality of any vacuum/siphon breakers associated with spent fuel or coolant inventory systems.

(3) IER 11-4, "Near-Term Actions to Address the Effects of an Extended Loss of All AC Power in Response to the Fukushima Daiichi Event," dated August 1, 2011

The earthquake and large tsunami that inundated the Fukushima Dai-ichi nuclear plant caused an extended loss of all AC power that resulted in emergency core cooling systems being unable to prevent fuel damage at three of the six units.

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 that would 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.

The recommendations call for the development of preplanned contingencies for protection from an extended loss of AC power and beyond station blackout events similar to those experienced at Fukushima Dai-ichi pending longer-term industry response. The recommendations also require stations to provide unit-specific information concerning 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. The four recommendations are listed below:

- 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. Included in this recommendation are implementing actions to address loss of AC power events (beyond station blackout) simultaneously at each unit of multiunit sites using the conditions described in the body of IER L1-11-4. Implement actions to improve operating margin that can be accomplished within the existing license.

  — Report the length of time the station can maintain critical safety functions listed in Recommendation 1 using existing installed equipment, even if less than 24 hours. Identify and report conditions that limit achieving 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 less than 24 hours. Identify and report conditions that limit achieving the 24-hour duration or longer durations. Describe and report the protective measures or measures that differentiate portable equipment from installed electrical AC power sources.

  — If enhancements or station upgrades are proposed for extending station ability to increase operating margin for extended loss of AC power events, include in your response the proposed upgrades and the expected margin improvement. The proposed upgrades will inform the industry working group process on methods and strategies recommended for broad industry consideration.

- 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 performing 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 installed DC batteries are depleted.

- 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 lasting 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.

- 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 or other communications infrastructures within 25 miles of the plant site.

(4) IER 13-10, "Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," dated March 28, 2013

In 2012, a team of INPO, WANO and U.S. industry personnel conducted an event review in Japan focusing on lessons learned for the broader nuclear industry. Tokyo Electric Power Company fully supported this review and provided full access to individuals and information available 1 year following the accident at Fukushima Dai-ichi. The lessons learned were summarized in a report made broadly available, and IER 13-10 was issued to provide recommendations for addressing the leadership, organizational, cultural, resource, and training issues that contributed to the event and that detracted from an effective emergency response. WANO issued a Significant Operating Experience Report containing similar recommendations.

IER 13-10 includes a broad set of recommendations, some with multiple parts. For brevity, the full text of the recommendations is not listed in this update to the Convention on Nuclear Safety Report for 2015. Instead, the following provides an overview of the recommended actions:

- Through senior manager and leader actions, foster development of a culture in which the staff recognizes that an extreme external event can occur and rigorous preparations must be made to respond to such an event.

- Establish enterprise and station risk management processes that consider nuclear risks, including those associated with changes in design basis assumptions for external events that could exceed the capability of installed equipment and accident response procedures.

- Implement emergency and accident response strategies for an extreme external event that provide multiple methods to restore and maintain safety functions, such as core cooling, emergency power, and containment integrity, using a defense-in-depth approach.

- Equip personnel responsible for performing emergency response duties with the required knowledge, skills, and proficiency to execute their roles.

- Staff organizations with sufficient personnel to respond effectively during initial stages of an extreme external event involving more than one unit at a multiunit site, and develop plans for staffing during long-duration events.

- Stage, maintain, test, secure, and programmatically control equipment needed for event response in a manner that protects from damage from the initiating event, supports timely deployment, and reduces likelihood of human error.

- Establish procedures and make preparations to enable the site organization to provide and receive assistance for mitigating a complex or long-duration emergency event.

Beginning in 2013 and continuing through 2015, an industry group coordinated by INPO developed training materials to assist utilities in preparing their organizations for beyond-design-basis events. These materials include case studies and instructor-led training focused on decisionmaking and decisionmaking under stress. The group also developed a guideline for establishing effective training for emergency response personnel. The guideline includes results of a job analysis that identified the needed knowledge and abilities required for each job function.

The INPO emergency plan and emergency response facilities were updated in early 2013 to better assist members in mobilizing the resources of the nuclear industry to provide assistance to a site experiencing an event. All INPO member utilities signed a mutual assistance agreement to provide resources during such an event, if requested. INPO conducts quarterly drills, most involving the Nuclear Energy Institute and the Electric Power Research Institute, to practice response actions. Some are conducted in conjunction with utilities during their regularly scheduled emergency preparedness drills. In July 2015, the industry response capability was demonstrated during a large-scale radiological release exercise conducted by Federal and State governmental agencies. This event, known as *Southern Exposure 2015*, was a fully integrated exercise that demonstrated the nation’s ability to effectively respond to a nuclear power plant event that resulted in widespread contamination to the surrounding community. The exercise was conducted in multiple phases over 5 days, beginning on July 21, 2015, and included time jumps to 14 days, 6 months, and 18 months postevent. Additional information can be found in Section 1.3.3 of Part 1 of this report.

Diverse and Flexible Coping Strategies (FLEX)

NEI and INPO worked with the U.S. nuclear industry to develop 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 a loss of normal access to the ultimate heat sink.

The objective is to establish an indefinite coping capability by relying upon installed equipment, onsite portable equipment, and prestaged offsite resources. The equipment ranges from diesel-driven pumps and electric generators to ventilation fans, hoses, fittings, cables and communications gear. The new equipment will be stored at diverse locations at the sites and protected to ensure that it can be used if other systems that compose a facility's multilayered safety strategy are compromised. This flexible approach builds on existing safety systems to protect against unforeseen events. 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, deploy onsite portable equipment.

- 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:

- Identify and plan for each site's offsite equipment needs
- Standardize interconnections
- Align with onsite coping strategies
- Determine required deployment times
- Designate offsite locations
- Establish 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 determine what coping equipment can be credited as coming from offsite sources. Procedures used to respond to FLEX address contacting the offsite sources. A standardized list of equipment connectors was developed to address interchangeability of the equipment. Each site is required to have one set of FLEX equipment onsite for each unit, plus one extra set. Therefore, these sites become a source of FLEX equipment for a site in such an event. 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 are two response centers, in Memphis and Phoenix, capable of delivering equipment to any site. The response centers are managed by a vendor, Strategic Alliance for FLEX Emergency Response. The Pooled Equipment Inventory Co. joined forces with AREVA to create this new company to develop and manage the response center program. Each response center has five sets of FLEX equipment: four sets to support sites and one set out of service for maintenance. Each center also has additional equipment specified by a site in their site-specific response center mobilization manual.

Each site has identified a staging area for delivery of the equipment. The response center will deliver the specified equipment to the staging area within 24 hours of being notified. Delivery will generally be made by air transport. Support will include equipment and may include equipment technicians to assist with setup and deployment. Qualified technicians from the 60 other facilities can be dispatched.

10. Conclusion

The U.S. commercial nuclear industry has made substantial, sustained, and quantifiable improvements in plant safety and performance during the 3 decades since Three Mile Island. 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, but it does take pride in its contribution to that success.

INPO also recognizes that the pursuit of excellence is a continuing journey. 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 2014 - 2018

The U.S. Nuclear Regulatory Commission (NRC) published NUREG-1614, Volume 6, "Strategic Plan: Fiscal Years 2014–2018" in August 2014. Appendix A to this report summarizes the key points of this plan.

Key Challenges

During the upcoming planning period, the NRC will face new challenges as it continues to operate in a dynamic environment. Key factors the agency has considered in developing this plan include the following:

- continued implementation of enhancements to improve nuclear safety based on insights arising from operating experience reviews and lessons learned from the 2011 nuclear accident at the Fukushima Dai-ichi nuclear facility in Japan
- continual learning and adaptation of the regulatory framework to address knowledge of and response to the specific hazards, uncertainties, and risks associated with each nuclear site
- continued readiness to review applications involving new technologies such as small modular reactors, medical isotope production facilities, and rapidly evolving digital instrumentation and control systems
- changes in the demographics, experience, and knowledge of the workforce
- continued awareness of and support to the development of nuclear safety and security regulations around the world
- changing economic conditions in the energy market affecting current and planned applications to construct and operate new nuclear facilities or decommission existing ones
- globalization of nuclear technology and the nuclear supply chain, driving the need for increased international engagement on the safe and secure use of radioactive material and the need for new oversight approaches, including ensuring that foreign components used in U.S. nuclear facilities are in compliance with NRC requirements
- continuous monitoring of the threat environment to ensure the security of facilities and accountability controls for radioactive materials.

To meet these challenges, the NRC must use its resources effectively and efficiently, enhance the regulatory framework as appropriate to address existing or emerging issues, and deploy effective and innovative strategies for maintaining staff competence and readiness. Even as the NRC works to address these challenges, the agency's mission and organizational values remain unchanged. The agency will remain a strong, independent, stable, and effective regulator that places the highest priority on ensuring the safety and security of the nuclear facilities and radioactive materials it regulates.

Key External Factors

The NRC's ability to achieve its strategic goals and their associated strategic objectives is influenced by many external factors, including industry operating experience, national priorities, the threat environment, legislation, Federal court litigation, market forces, and resource availability. The NRC will strengthen its ability to manage change and maintain its readiness to respond promptly to any agency priority shifts necessitated by factors that are beyond its control. The agency will also make efforts to influence those factors that enable the achievement of its strategic objectives.

External Factors Affecting Safety Objective 1: Prevent and mitigate accidents and ensure radiation safety.

- Market Pressures on Operating Plants and License Applications. Market forces result in pressures for licensees to reduce operating costs. As a result, the NRC needs to be prepared for changes in workload and to address potential shutdowns of facilities before license expiration and to continue to ensure that oversight programs identify degrading facility safety and security performance. Conversely, the lower capital costs of small modular reactors (under 300 megawatts) may offer industry a more attractive option to add new capacity. Several entities are seeking to submit license applications for small modular reactors in the next several years. The U.S. Department of Energy (DOE) is funding a program "to design, certify and help commercialize innovative small modular reactors (SMRs) in the United States." Therefore, the NRC is developing a licensing framework for these as well as other advanced reactors.

- Significant Operating Incident at a Non-U.S. Nuclear Facility. A significant incident at a nuclear facility outside the United States could cause the agency to reassess its safety and security requirements, which could change the agency's focus on some initiatives related to its objectives until the situation stabilizes.

- Significant Operating Incident at a Domestic Nuclear Facility. A significant incident at a U.S. 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 objectives until the situation stabilizes. Because the NRC's stakeholders are highly sensitive to many issues regarding the use of radioactive materials, even events of relatively minor safety significance could potentially require a response that consumes considerable agency resources.

- International Nuclear Standards Developments. International organizations, such as the International Atomic Energy Agency, will continue to develop and issue standards and guidance affecting global commitments to nuclear safety and security. To ensure that the best results are achieved both domestically and internationally, the NRC needs to proactively engage in these international initiatives and to provide leadership in a cooperative and collegial manner.

- International Treaties and Conventions. As part of the international response to lessons learned from the Fukushima Dai-ichi nuclear accident in Japan, the international nuclear regulatory community is reviewing the Convention on Nuclear Safety. As one of the contracting parties to the Convention, the NRC is a member of the working group that is reviewing the Convention. Likewise, the NRC participates in the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

- Globalization of the Nuclear Technology and the Nuclear Supply Chain. Components for nuclear facilities are increasingly manufactured overseas, resulting in challenges of providing effective oversight to ensure that these components are in compliance with NRC requirements. In addition, the continuing globalization of nuclear technology is driving the need for increasing international engagement on the safe use of radioactive material.

External Factors Affecting Security Objective 1: Ensure protection of nuclear facilities and radioactive materials.

- Significant Terrorist Incident. A sector-specific credible threat or actual significant terrorist incident anywhere in the United States would result in the Department of Homeland Security raising the threat level under the National Terrorism Advisory System. In turn, the NRC would similarly elevate the oversight and response stance for NRC-regulated facilities and licensees. Potentially, new or revised security requirements or other policy decisions might affect the NRC, its partners, and the regulated community. In a similar fashion, a significant terrorist incident at a nuclear facility or activity anywhere in the world would need to be assessed domestically and potentially lead to a modification of existing security requirements for NRC-regulated facilities and licensees.

- International Treaties and Conventions. The ratification by the United States of international instruments related to the security of nuclear facilities or radioactive materials could potentially impose binding provisions on the Nation and the corresponding governmental agencies, such as the NRC and DOE.

- Globalization of Nuclear Technology. The continuing globalization of nuclear technology is driving the need for increased international engagement on the secure use of radioactive material.

- Legislative and Executive-Branch Initiatives. Congressional and Executive Branch initiatives concerning cyber security could affect the NRC's regulatory framework for nuclear security. If the NRC were to become concerned about an aspect of a bill or policy initiative that had been introduced, the staff would consult the Commission to develop a strategy for making such concerns known.

External Factors Affecting Security Objective 2: Ensure protection of classified and Safeguards Information.

- Lost, Misplaced, Intercepted, or Delayed Information. With the increased use of mobile devices and alternative storage options, the introduction of new communication technologies, and the increased use of telecommunication, there is a heightened risk that sensitive information held by the NRC or its licensees can be lost, misplaced, or intercepted and fall into the hands of unauthorized persons.

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 2015; this report can be found on the agency's public Web site.

The Fiscal Year (FY) 2016 management and performance challenges are directly related to the NRC's mission areas (i.e., commercial nuclear reactors and nuclear materials), security, information technology and information management, financial programs, and administrative functions. The agency's work in these areas indicates that while program improvements are needed, the NRC is continually making progress to address the Inspector General's recommendations and improve the efficiency and effectiveness of its programs. These challenges represent what the Inspector General considers to be inherent and continuing program challenges relative to maintaining effective and efficient oversight and internal controls. As a result, it is likely they will continue to be challenges from year to year. Challenges do not necessarily equate to problems. In the 2015 report, the Inspector General identified the six management challenges described below to be the most serious as of October 1, 2015.

Challenge 1: Regulation of nuclear reactor safety programs

The NRC is responsible for maintaining an established regulatory framework for the safe and secure use of civilian nuclear reactors, including commercial nuclear power plants as well as research, test, and training reactors. As of October 1, 2015, there are 99 nuclear power plants licensed to operate in the United States, which generate about 20 percent of the Nation's electrical use, as well as 5 plants under construction (i.e., Vogtle Units 3 and 4; Summer Units 2 and 3; Watts Bar Unit 2). There are also 31 licensed research and test reactors. The NRC's regulatory oversight responsibilities in the reactor arena include developing policy and rulemaking, licensing and inspecting reactors, licensing reactor operators, and enforcing regulations. The agency implements the nuclear reactor safety program with approximately 77 percent ($810 million) of its total budget authority and 76 percent (2,900 full-time equivalent employees) of its total staff. Thus, it is of paramount importance that the agency implements these programs as effectively and efficiently as possible.

Challenge 2: Regulation of nuclear materials and radioactive waste programs

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, conversion and enrichment activities; fuel fabrication and development; and high-level and low-level radioactive waste. 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. 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 the environment, and compatible with the NRC's program. The States that enter into an agreement assuming this regulatory authority from the NRC are called

Agreement States. Currently, there are 37 Agreement States.

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 waste material remaining after spent fuel is reprocessed. Because radioactive waste becomes harmless only through decay (which may take hundreds of thousands of years for high-level waste), the material must be stored and ultimately disposed of in a manner that provides adequate protection of the public for a very long time.

Low-level radioactive waste is typically produced at nuclear power reactors, hospitals, research facilities, and clinics from the use of nuclear materials for industrial and medical purposes. The NRC regulates the management, storage, and disposal of radioactive waste produced as a result of NRC-licensed activities. Low-level radioactive waste includes contaminated protective clothing, equipment and tools, medical supplies, and laboratory animal tissues.

Challenge 3: Management of security over internal infrastructure (personnel, physical, and cyber security) and nuclear security

The NRC must remain vigilant with regard to the security of its infrastructure and that of nuclear facilities and nuclear materials. The NRC 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 oversight of protecting nuclear facilities and materials, the sharing of sensitive information, as well as emergency preparedness and incident response.

Challenge 4: Management of information technology and information management

Technology advances rapidly. New technologies such as cloud, virtualization, and mobility are tools that can be implemented. The challenge is deciding which of these new technologies will work to the best interest of the NRC now. The mission of the NRC's information technology/information management program is to manage information and employ information technology to enhance information access and strengthen agency performance. The most important goal of the NRC's information technology/information management program is effective information access— enabling both NRC staff and the public to quickly and easily obtain the information they need. This goal reflects the NRC's commitment to openness and is essential for effective agency operations.

Challenge 5: Management of financial programs

The NRC is required by the Omnibus Budget Reconciliation Act of 1990 to collect fees totaling approximately 90 percent of its annual budget authority. The agency's budget authority for FYs 2013 and 2014 was $985.6 million and $1,055.9 million, respectively. The NRC estimated that $859.6 million for FY 2013 and $916.7 million for FY 2014 should be recovered from invoiced fees. The NRC is required to establish a schedule of charges that fairly and equitably assess the fees to license holders and license applicants. In recent years, multiple external stakeholders have questioned the NRC's budget and fees structure. To maintain transparency, the NRC must continue to implement solid internal controls over financial management and reporting.

Challenge 6: Management of administrative functions

The NRC should continue exploring ways to reduce administrative inefficiencies while maintaining the appropriate corporate support to carry out agency operations. During FY 2015, NRC workforce totaled approximately 3,700 staff positions. To support the agency's technical staff, the NRC provides corporate support services, such as contract support and multiple human resource programs. Although the NRC has implemented multiple programs to support agency staff, the NRC continues to operate in a Federal Government environment of stagnant or reduced agency budgets, and increasing pressure to reduce corporate support costs. Because of this, the agency needs to have an adequate balance between administrative functions and technical needs. In addition, the NRC must be able to effectively recruit, train and transfer knowledge to new hires. This includes maintaining up-to-date guidance to effectively transfer knowledge and train current staff.

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APPENDIX D
U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: U.S. Nuclear Regulatory Commission NUREG-1350, Volume 27, “2015-2016 Information Digest,” August 2015.

Subsequent to the issuance of NUREG-1350, Volume 27, on October 22, 2015, the NRC issued the operating license for Watts Bar Unit 2, which is also listed in the table below.

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	3645	07/88	10/26
Braidwood 2 - Exelon Corp., Exelon Generation Corporation, LLC	PWR	3645	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	3645	09/85	10/44
Byron 2 – Exelon Corp., Exelon Generation Corporation, LLC	PWR	3645	08/87	11/46
Callaway – AmerenUE, Union Electric Company	PWR	3565	12/84	10/44
Calvert Cliffs 1 - Constellation Energy	PWR	2737	05/75	07/34

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
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
Davis-Besse - FirstEnergy Nuclear Operating Co.	PWR	2817	07/78	04/37
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
Fermi 2 – The Detroit Edison Co.	BWR	3486	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	4408	07/85	11/24
H.B. Robinson 2 - Carolina Power & Light Co.	PWR	2339	03/71	07/30

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
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
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	10/44
Limerick 2- Exelon Corporation, Exelon Generation Co., LLC	BWR	3515	01/90	06/49
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
Millstone 3 – Dominion Nuclear Connecticut, Inc., Dominion Generation	PWR	3650	04/86	11/45
Monticello - Nuclear Management Co.	BWR	2004	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

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
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
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

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
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
Seabrook 1 - FPL Energy Seabrook, LLC	PWR	3648	08/90	03/30
Sequoyah 1 - Tennessee Valley Authority	PWR	3455	07/81	09/40
Sequoyah 2 - Tennessee Valley Authority	PWR	3455	06/82	09/41
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	3020	12/76	03/36
St. Lucie 2 - Florida Power & Light Co.	PWR	3020	08/83	04/43
St. Lucie 1 - Florida Power & Light Co.	PWR	2587	12/72	05/32
St. Lucie 2 - Florida Power & Light Co.	PWR	2587	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	2644	12/72	07/32
Turkey Point 4 - Florida Power & Light Co.	PWR	2644	09/73	04/33

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime	
V.C. Summer - South Carolina Electric & Gas Co.	PWR	2900	01/84	08/42
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
Watts Bar 2 - Tennessee Valley Authority	PWR	3411	10/15	10/55
Wolf Creek 1 - Wolf Creek Nuclear Operating Corporation	PWR	3565	09/85	03/45

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September 2013 August 2016

8 PERFORMING ORGANIZATION NAME AND ADDRESS (If NRC, provide Division, Office or Region, U S Nuclear Regulatory Commission, and mailing address, if contractor, provide name and mailing address)
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9 SPONSORING ORGANIZATION NAME AND ADDRESS (If NRC, type "Same as above" if contractor, provide NRC Division, Office or Region, U S Nuclear Regulatory Commission, and mailing address)
Same as above

10 SUPPLEMENTARY NOTES
This reports is an update to NUREG-1650, Revision 5

11 ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission has prepared Revision 6 to NUREG-1650, "The United States of America Seventh National Report for the Convention on Nuclear Safety for submission for peer review at the seventh review meeting of the Convention on Nuclear Safety to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2017 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 also addresses the principles of the Vienna Declaration adopted by the Contracting Parties in February 2015.

Similar to the U.S. National Report issued in 2013, this revised document includes 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

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers locating the report)
Convention on Nuclear Safety CNS, treaties, nuclear plants, programs, challenges, legislation, regulation, reactor oversight, public participation, probabilistic risk assessment, PRA, performance-based, risk-based, quality safety siting, design, construction, operations, new reactors, radiation protection, emergency preparedness, financial, human factors, Institute of Nuclear Power Operations, INPO, peer review, Project AIM, Vienna Declaration, Fukushima.

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