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NUREG-1650
Revision 7

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

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

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ABSTRACT

The U.S. Nuclear Regulatory Commission has prepared Revision 7 to NUREG-1650, “The United States of America Eighth National Report for the Convention on Nuclear Safety,” for submission for peer review at the eighth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2020. This report addresses the safety of land-based commercial nuclear power plants in the United States (U.S.). It demonstrates how the U.S. Government achieves and maintains a high level of nuclear safety worldwide by enhancing national measures and international cooperation and by meeting the obligations of all the articles established by the Convention. These articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee, the priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation. 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 2016, 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.

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PART 3 THE ROLE OF THE INSTITUTE OF NUCLEAR POWER OPERATIONS IN SUPPORTING THE UNITED STATES COMMERCIAL NUCLEAR POWER INDUSTRY'S FOCUS ON NUCLEAR SAFETY

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has prepared Revision 7 to NUREG-1650, "The United States of America Eighth National Report for the Convention on Nuclear Safety," for submission for peer review at the eighth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2020. 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 seventh review meeting in March 2017 and discusses challenges and issues that have arisen since that time. The seventh review meeting identified the following four U.S. challenges:

(1) establishment of the acceptance criteria for operation beyond 60 years
(2) clarifying the backfitting guidance and implementation criteria
(3) changes in the demographics, experience, and knowledge of the staff
(4) ensuring continuity during the oversight transition from plant construction to operation

This report discusses the status of safety issues raised in the seventh U.S. National Report, including baffle-former bolts, digital instrumentation and control systems, open phase conditions in electric power system, Project Aim, risk-informing regulations and processes, spent fuel pool neutron-absorbing materials, staff readiness to transition plants from construction to operations, staff readiness to transition plants from operation to decommissioning, and subsequent license renewal. The report also addresses the following safety and regulatory issues that have needed significant attention since 2016:

- accident tolerant fuel
- changes to the Reactor Oversight Process
- clarifying the backfit process
- digital instrumentation and control systems
- proposed rulemaking on emergency preparedness for small modular reactors and other new technologies
- risk-informed decisionmaking
- subsequent license renewal challenges
- transformation at the U.S. Nuclear Regulatory Commission

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.

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ABBREVIATIONS AND ACRONYMS

ABWR	advanced boiling-water reactor
AC	alternating current
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
BTP	branch technical position
BWR	boiling-water reactor
CEO	chief executive officer
CFR	*Code of Federal Regulations*
CNS	Convention on Nuclear Safety
CNSC	Canadian Nuclear Safety Commission
CNSNS	National Nuclear Safety and Safeguards Commission of the United Mexican States
CRGR	Committee to Review Generic Requirements
dc	direct current
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
GSI	generic safety issue
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 Report
IMC	Inspection Manual chapter
IN	information notice
INPO	Institute of Nuclear Power Operations
IP	inspection procedure

IPSR	INPO Performance Summary Report
IRRS	Integrated Regulatory Review Service
ISG	interim staff guidance
ITAAC	inspection(s), test(s), analysis (analyses), and acceptance criterion (criteria)
LER	licensee event report
LOCA	loss-of-coolant accident
MD	management directive
MRP	Materials Reliability Program (EPRI)
MWe	megawatt electric
MWt	megawatt thermal
NANTeL	National Academy for Nuclear Training e-Learning
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NIMS	National Incident Management System
NQA	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
OIG	Office of the Inspector General
OSART	Operational Safety Assessment Review Team
POs&Cs	performance objectives and criteria
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RG	regulatory guide
RIDM	risk-informed decisionmaking
RISC	Risk Informed Steering Committee
RIS	regulatory issue summary
SAMGs	severe accident management guidelines
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluation
SAT	systems approach to training
SEE-IN	Significant Event Evaluation and Information Network
SFP	spent fuel pool
SRM	staff requirements memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
SSC	structure, system, and component
Sv	Sievert
SWP	Strategic Workforce Planning
TSTF	Technical Specification Task Force
U.S.	United States
U.S. APWR	U.S. Advanced Pressurized-Water Reactor
WANO	World Association of Nuclear Operators
WCAP	Westinghouse Commercial Atomic Power

PART 1
Introduction and Summary

1 INTRODUCTION

The introduction describes the purpose and structure of the "United States of America Eighth National Report for the Convention on Nuclear Safety," and provides a listing of changes.

1.1 Purpose and Structure of This Report

The United States of America is submitting this updated report for peer review to the eighth review meeting of the contracting parties to the Convention on Nuclear Safety (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 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 (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 seventh U.S. National Report published in 2016 and is designed to be a standalone document. Therefore, this report duplicates some of the information presented in the 2016 report. To facilitate peer review, Table 1 includes a summary of the main changes to the report. Table 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.

Similar to the 2016 report, Part 3 of this document includes a contribution by the Institute of Nuclear Power Operations (INPO) describing work done by the U.S. nuclear industry to ensure

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

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 appendices that contain references and list 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 (http://www.nrc.gov).

1.2 Changes to the Seventh U.S. National Report

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

Table 1 Listing of Changes to the Seventh U.S. National Report

Report Section		Change
Abstract		Updated to add discussion about the seventh CNS
Executive Summary		Updated to add discussion about the seventh CNS
PART 1		
Section 1	INTRODUCTION	Renumbered and updated to add discussion about the seventh CNS
1.1	Purpose and Structure of This Report	Updated to add discussion about the seventh CNS
1.2	Changes to the Seventh U.S. National Report	Updated table
Section 2	SUMMARY	Renumbered and updated to add discussion about the seventh CNS
2.1	The U.S. Policy toward Nuclear Activities	No changes
2.1.1	Regulatory Body Organizational Values	Editorial changes only
2.1.2	Regulatory Body Challenges	Updated to add discussion on most recent NRC Strategic Plan and Inspector General report
2.2	National Nuclear Programs	Editorial changes only
2.2.1	Reactor Oversight Process	Moved self-assessment discussion to Article 6
2.2.2	License Renewal	Updated discussion about units entering the period of extended operation
2.2.3	Power Uprates	Editorial changes only
2.2.4	New Reactor Licensing	Updated number of applications under review and discussion on international activities

Report Section		Change
2.3	Safety and Regulatory Issues, and Regulatory Accomplishments	Editorial changes only
2.3.1	Safety and Regulatory Issues Discussed in the Seventh U.S. National Report	Updated to discuss current status and activities conducted in the last 3 years
2.3.2	Current Safety and Regulatory Issues	Completely updated to address new topics
2.3.3	Major Regulatory Accomplishments	Completely updated to address new topics
2.4	International Peer Reviews and Missions	Editorial changes only
2.4.1	Convention on Nuclear Safety	Updated to (1) include results from seventh CNS peer review and country review report findings, (2) summarize implementation of the Vienna Declaration on Nuclear Safety principles, and (3) address areas of focus for the eighth CNS
2.4.2	Integrated Regulatory Review Service	Updated to include reference to mission results
2.4.3	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 generic communications
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 shutdown
6.3	Regulatory Processes and Programs	No change
6.3.1	Reactor Licensing	Updated to include information on applications under review
6.3.2	Reactor Oversight Process	Updated to discuss current plant performance status and transformation activities
6.3.3	Industry Trends Program	Updated to confirm discontinuation of the program
6.3.4	Accident Sequence Precursor Program	Updated to discuss issuance of annual report
6.3.5	Operating Experience Program	Updated to discuss the development of the center of expertise
6.3.6	Generic Issues Program	Added reference to office instruction
6.3.7	Rulemaking	Updated to add discussion on opportunities for public participation and openness
6.3.8	Fire Protection Regulation Program	Updated to discuss issuance of guidance document
6.3.9	Decommissioning	Updated to discuss decommissioning rulemaking
6.3.10	Reactor Safety Research Program	Editorial changes only
6.3.11	Generic Communications and Orders	Eliminated public participation discussion because of duplication. Relevant discussion was moved and integrated into Articles 6, 8, and 9. New section discusses generic communications and orders.
6.4	Vienna Declaration on Nuclear Safety	Updated to make reference to summary section
Article 7	LEGISLATIVE AND REGULATORY FRAMEWORK	Editorial changes only
7.1	Legislative and Regulatory Framework	Editorial changes only
7.2	Provisions of the Legislative and Regulatory Framework	No changes
7.2.1	National Safety Requirements and Regulations	Editorial changes only
7.2.2	Licensing of Nuclear Installations	Editorial changes only

Report Section		Change
7.2.3	Inspection and Assessment	No changes
7.2.4	Enforcement	Updated discussion on the maximum civil penalty amount
Article 8	REGULATORY BODY	Editorial changes only
8.1	The Regulatory Body	Editorial changes only
8.1.1	Mandate	No changes
8.1.2	Authority and Responsibilities	No changes
8.1.2.1	Scope of Authority	Editorial changes only
8.1.2.2	The NRC as an Independent Regulatory Agency	Expanded discussion on NRC’s authority
8.1.3	Structure of the Regulatory Body	Editorial changes only
8.1.3.1	The Commission	Editorial changes only
8.1.3.2	Component Offices of the Commission	Updates to more accurately reflect the roles and responsibilities of the offices.
8.1.3.3	Offices of the Executive Director for Operations	Updated to reflect that there was no office reorganization during this reporting period. Remaining changes more accurately reflect the roles and responsibilities of the offices.
8.1.3.4	Advisory Committees	Updates to more accurately reflect the roles and responsibilities of the committees
8.1.3.5	Atomic Safety and Licensing Board Panel	Updates to more accurately reflect the roles and responsibilities of the committees
8.1.3.6	Office of the Inspector General	No changes
8.1.4	Position of the NRC in the Governmental Structure	No changes
8.1.4.1	Executive Branch	Updated to more accurately reflect the roles and responsibilities of the agencies
8.1.4.2	The States (i.e., of the United States)	Editorial changes only
8.1.4.3	Congress	Editorial changes only
8.1.5	International Responsibilities and Activities	Updated throughout. Provided list of missions supported in the last reporting period.
8.1.5.1	International Standards	Editorial changes only
8.1.5.2	Integrated Regulatory Review Service Mission	Summarized
8.1.5.3	Operational Safety Assessment Review Teams	Updated to add discussion on recent 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 2019
8.1.6.2	Human Resources	Updated throughout, including discussion on Strategic Workforce Planning efforts
8.1.7	Openness and Transparency	Updated throughout, including most recent numbers associated with public outreach activities
8.2	Separation of Functions of the Regulatory Body from Those of Bodies Promoting Nuclear Energy	Editorial changes only
8.3	Ethics Rules Applying to NRC Employees and Former Employees	Updated throughout
Article 9	RESPONSIBILITY OF THE LICENSE HOLDER	Editorial changes only
9.1	Introduction	No changes
9.2	The Licensee’s Primary Responsibility for Safety	Editorial changes and updated references.

Report Section		Change
9.3	Mechanisms to Enforce Licensee's Responsibilities to Maintain Safety	Retitled and restructured to include three subsections
9.3.1	Enforcement Program	Updated discussion on available enforcement sanctions. Updated table of enforcement actions.
9.3.2	NRC Petition for Enforcement Process	New section
9.3.3	Allegation Program	Updated to provide new references
9.4	Openness and Transparency	Editorial changes only
9.5	Financial and Human Resources	Restructured to include a standalone discussion
9.5.1	Financial Resources	Editorial changes only
9.5.2	Human Resources	Updated references
Article 10	PRIORITY TO SAFETY	Updated to add references to policy statements
10.1	Background	Editorial changes only
10.2	Probabilistic Risk Assessment Policy	Editorial changes only
10.2.1	Applications of Probabilistic Risk Assessment	Renumbered and restructured. Consolidates and summarizes all risk assessment sections in Sections 10.2 and 10.3 of the previous report.
10.2.2	Level 3 Probabilistic Risk Assessment Project	Renumbered. Updated to provide latest project status.
10.3	Safety Culture	Renumbered. No additional changes.
10.3.1	Safety Culture Policy Statement	Renumbered. Editorial changes only.
10.3.2	NRC Monitoring of Licensee Safety Culture	Renumbered. No additional changes.
10.3.2.1	Background	Renumbered. Editorial changes only.
10.3.2.2	Enhanced Reactor Oversight Process	Renumbered. Editorial changes only.
10.3.3	NRC Safety Culture	Renumbered. Updated references.
10.4	Managing the Safety and Security Interface	Renumbered. Updated references.
Article 11	FINANCIAL AND HUMAN RESOURCES	Editorial changes only
11.1	Financial Resources	Updated references
11.1.1	Financial Qualifications for Construction and Operations	Updated to add discussion on rulemaking activities
11.1.1.1	Construction Permit Reviews	Updated references
11.1.1.2	Operating License Reviews	Updated to clarify the NRC's authority
11.1.1.3	Combined License Application Reviews	Updated references
11.1.1.4	Reviews of License Transfers	Renumbered. Updated to add discussion on information contained in applications.
11.1.2	Financial Assurance for Decommissioning	Minor editorial change in section title. Updated to add discussion on rulemaking activities.
11.1.3	Financial Protection Program for Liability Claims Arising from Nuclear Incidents	Minor editorial change in the section title. Updated the "retrospective premium pool" requirements and rulemaking activities.
11.1.4	Insurance Program for Onsite Property Damages Arising from Nuclear Incidents	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	Updated to reflect experience in the last reporting cycle
Article 12	HUMAN FACTORS	Editorial changes only
12.1	Overview of Regulatory Requirements	Article 12 was restructured and streamlined to better align with IAEA reporting guidelines. Section discusses regulatory requirements.

Report Section		Change
12.2	Regulatory Review and Control Activities	Article 12 was restructured and streamlined to better align with IAEA reporting guidelines
12.2.1	Nuclear Power Plant Design and Modifications and Operator Actions	Updated references
12.2.2	Organizational Issues	New section
12.2.3	Emergency Operating Procedures and Plant Procedures	Updated status on post-Fukushima activities and information on mitigating strategies rule
12.2.4	Shift Staffing	Added discussion on small modular reactors and new technologies
12.2.5	Human Performance in the Reactor Oversight Process	Updated discussion on inspection procedures and recent experience
12.2.6	Human Factors Information System	New section
12.2.7	Fitness for Duty	Updated references
12.3	Licensee Human Factors Program	New section
12.4	Feedback and Experience	New section
12.4.1	Human Factors Associated with Digital Instrumentation and Control	New section
12.4.2	Human Performance in Decommissioning Activities	New section
12.4.3	Human Performance Research	New section
Article 13	QUALITY ASSURANCE	Editorial changes only
13.1	Background	Editorial changes only
13.2	Regulatory Policy and Requirements	Editorial changes only
13.2.1	Appendix A to 10 CFR Part 50	Editorial changes only
13.2.2	Appendix B to 10 CFR Part 50	Editorial changes only
13.2.3	Approaches for Adopting More Widely Accepted International Quality Standards	Editorial changes only
13.3	Quality Assurance Regulatory Guidance	Editorial changes only
13.3.1	Guidance for Staff Reviews for Licensing	Editorial changes only
13.3.2	Guidance for Design and Construction Activities	Updated references
13.3.3	Guidance for Operational Activities	Editorial changes only
13.4	Quality Assurance Programs	Editorial changes only
13.5	Quality Assurance Audits Performed by Licensees	Editorial changes only
13.5.1	Audits of Vendors and Suppliers	No changes
13.6	Vendor Inspection Program	Updated references
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	Editorial changes only
14.1.2	Maintaining the Licensing Basis	Editorial changes only
14.1.2.1	Governing Documents and Process	Updated references
14.1.3	Power Uprates	No changes
14.1.3.1	Governing Documents and Process	Updated references
14.1.3.2	Experience	Updated discussion on power uprates approved in the last reporting cycle
14.1.4	License Renewal	No changes

Report Section		Change
14.1.4.1	Governing Documents and Process	Updated references and discussion associated with subsequent license renewals and operation beyond 60 years
14.1.4.2	Experience	Updated discussion about license renewals approved in the last reporting cycle
14.1.4.3	Operating beyond 60 Years	Updated throughout
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	Updated to clarify scope of license renewal and to highlight the role of the Maintenance Rule in monitoring active components
14.1.5.2	The Backfitting Process: Timely Imposition of New Requirements	Updated to clarify the process and scope. Also clarified role of the Committee to Review Generic Requirements.
14.1.5.3	License Renewal Confirms Safety of Plants	Updated references
14.1.5.4	Risk-Informed Regulation and the Reactor Oversight Process	Updated to clarify risk-informed approach
14.1.5.5	Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations	Editorial changes only
14.1.5.6	The NRC's Regulatory Process Compared with International Safety Reviews	Updated references
14.2	Verification by Analysis, Surveillance, Testing, and Inspection	Updated to add examples of the 10 CFR 50.54(f) information request process
14.3	Vienna Declaration on Nuclear Safety	Updated to refer to summary section 2.4.1
Article 15	RADIATION PROTECTION	Editorial changes only
15.1	Overview of Regulatory Requirements and Authority	Section renamed. Multiple sections merged and restructured to better align with IAEA reporting guidance.
15.2	Regulatory Framework and Expectations	Section renamed. Multiple sections merged and restructured to better align with IAEA reporting guidance. Added discussion on 10 CFR Part 71 requirements for radioactive material transportation.
15.3	Radiation Protection Activities and Control of Radiation Exposure	Section renamed, renumbered, and restructured to better align with IAEA reporting guidance. Updated reference to effluent reports.
15.3.1	Control of Radiation Exposure of Occupational Workers	Section renumbered. Updated collective doses
15.3.2	Control of Radiation Exposure of Members of the Public	Section renumbered. Updated references and exposure limits.
Article 16	EMERGENCY PREPAREDNESS	Editorial changes only
16.1	Emergency Plans and Programs	New section title. Section restructured to better align with the IAEA reporting guidelines.
16.1.1	Background and Overview of Regulatory Requirements	Renumbered. Updated section title. Updated section to discuss post-Fukushima enhancements.
16.1.2	National Response to an Emergency	Renumbered. Updated references and Tribal responsibilities.
16.1.2.1	Federal Response	Renumbered. Updated responsibilities.
16.1.2.2	Licensee, State, Tribal, and Local Response	Renumbered. Updated to add Tribal response.

Report Section		Change
16.1.2.3	NRC Response	Renumbered. Updated to clarify site team response
16.1.2.4	Aspects of Security that Support Response	Renumbered. Simplified discussion of changes after the September 2001 events
16.1.3	Implementation of Emergency Preparedness Measures	New section title
16.1.3.1	Emergency Classification System and Emergency Action Levels	Renumbered. Summarized section and updated references and severe accidents discussion.
16.1.3.2	Offsite Emergency Planning and Preparedness	Renumbered. Updated references.
16.1.3.3	Emergency Preparedness Facilities	New section
16.1.3.4	Recommendations for Protective Action in Severe Accidents	Renumbered. Updated references and discussion on potassium iodine.
16.1.4	Emergency Response Exercises	Renumbered. Updated references.
16.1.5	Regulatory Review and Inspection Practices	Renumbered. Updated title.
16.2	Communications Activities	New section title
16.2.1	Communications with Neighboring States and International Arrangements	Renumbered. Updated information on agreements and added discussion on observation of exercises.
16.2.2	Communications with the Public	Renumbered. Updated references.
Article 17	SITING	Editorial changes only
17.1	Background	Editorial changes only
17.2	Safety Elements of Siting	No changes
17.2.1	Background	Updated references
17.2.2	Assessments of Non-seismic Aspects of Siting	Updated references
17.2.3	Assessments of Seismic and Geological Aspects of Siting	Updated references
17.2.4	Assessments of Radiological Consequences from Postulated Accidents	Updated references
17.3	Environmental Protection Elements of Siting	No changes
17.3.1	Governing Documents and Process	Removed old references and summarized section
17.3.2	Other Considerations for Environmental Reviews	Removed old references and summarized section
17.4	Reevaluation of Site-Related Factors	Updated references
17.5	Consultation with Other Contracting Parties To Be Affected by the Installation	Updated references to Tribal Policy Statement
17.6	Vienna Declaration on Nuclear Safety	Updated to refer to summary section
Article 18	DESIGN AND CONSTRUCTION	Editorial changes only
18.1	Implementation of Defense-in-Depth	New section title. Section restructured to better align with the IAEA reporting guidelines.
18.1.1	Overview of Regulatory Requirements and Governing Documents	New section of references
18.1.2	Application of the Defense-in-Depth Philosophy	Renumbered. Updated to add a simplified discussion of the application of the philosophy.
18.1.3	Regulatory Review and Control Activities	Renumbered. Updated references and added discussion on lessons from the new construction.

Report Section		Change
18.1.4	Experience and Implementation of Defense-in-Depth Measures	Updated title and added discussion on implementation of lessons from the Fukushima accident
18.2	Technologies Proven by Experience or Qualified by Testing or Analysis	Updated discussion on topical reports
18.3	Design for Reliable, Stable, and Easily Manageable Operation	No changes
18.3.1	Governing Documents and Process	Updated references
18.3.2	Experience	Updated to reflect current experience
18.3.2.1	Human Factors Engineering	Updated to reflect current experience
18.3.2.2	Digital Instrumentation and Controls	Updated to reflect current experience
18.3.2.3	Cybersecurity	Updated to reflect current experience and references
18.4	New Reactor Construction Experience Program	Updated discussion on center of expertise
18.5	Vienna Declaration on Nuclear Safety	Renumbered. Updated to refer to summary section.
Article 19	OPERATION	Editorial changes only
19.1	Initial Authorization to Operate	Updated to summarize discussion and include applications approval status
19.2	Definition and Revision of Operational Limits and Conditions	Updated discussion on technical specifications
19.3	Approved Procedures	Editorial changes only
19.4	Procedures for Responding to Anticipated Operational Occurrences and Accidents	Updated discussion on mitigating strategies rulemaking
19.5	Availability of Engineering and Technical Support	Updated references
19.6	Incident Reporting	Updated references, discussion on significant reactor events, and petition for rulemaking
19.7	Programs To Collect and Analyze Operating Experience	Updated references and discussion on center of expertise
19.8	Radioactive Waste	Updated references, waste amounts, and repository discussion
19.9	Vienna Declaration on Nuclear Safety	Updated to refer to summary section
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		Updated throughout
APPENDICES		
APPENDIX A REFERENCES		Renumbered and updated. Appendix A in the previous report included a summary of the NRC Strategic Plan. This is now summarized in Section 2 of this report.
APPENDIX B U.S. COMMERCIAL NUCLEAR POWER REACTORS		Renumbered and updated. Appendix B in the previous report included a summary of the NRC challenges. This is now summarized in Section 2.

2 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, includes an update on important safety and regulatory issues identified in the previous National Report, and addresses those safety and regulatory issues that have arisen and regulatory accomplishments since the last National Report was issued (see NUREG-1650, Revision 6, "The United States of America Seventh National Report for the Convention on Nuclear Safety," issued in August 2016). Lastly, this section summarizes the results of international peer reviews and missions.

2.1 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.

2.1.1 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 guide NRC regulatory activities. These principles focus on ensuring safety and security while appropriately balancing the interests of stakeholders, including licensees; state, local, and tribal governments; nongovernmental organizations; and the public. These principles are independence, efficiency, clarity, reliability, and openness. The NRC's 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.

Because the NRC views nuclear regulation as a service to the public, this function 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, reflects the agency's long history of, and commitment to, openness with the public and transparency in the regulatory process. The agency's goal of ensuring openness explicitly recognizes that the public must be informed about, and have a reasonable opportunity to participate meaningfully in, the regulatory process. Except for certain classes of information, including proprietary information, security-related information, pre-decisional 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. The NRC also has embraced social media as an important tool for reaching a wider public audience. As a result, much of the information about nuclear activities and the national policy about them is available to everyone.

2.1.2 Regulatory Body Challenges

The NRC identified major challenges for the future in NUREG-1614, Volume 7, "Strategic Plan: Fiscal Years 2018-2022," dated February 2018. Many external factors influence the ability of the NRC to achieve its strategic goals and the associated strategic objectives. These factors include:

- industry operating experience
- national priorities
- a significant incident at a domestic or non-U.S. nuclear facility
- the security and threat environment
- legislation
- Federal court litigation
- market forces
- new technologies
- resource availability

The NRC continues to strengthen its ability to anticipate and respond promptly to shifts in agency priorities necessitated by these factors.

By law, the Inspector General of each Federal agency (as discussed under Article 8) identifies the agency's most serious management and performance challenges. These challenges do not necessarily equate to problems; rather, they are areas of continuing important focus for NRC management and staff. The "Inspector General's Assessment of the Most Serious Management and Performance Challenges Facing the Nuclear Regulatory Commission (NRC) in FY 2019" (OIG-19-A-01), dated October 28, 2018, discusses what the NRC's Inspector General considers to be mission critical areas or programs that have the potential for a perennial weakness or vulnerability that, without substantial management attention, would seriously impact agency operations or strategic goals. The fiscal year (FY) 2019 management and performance challenges are the following:

- regulation of nuclear reactor safety and security programs
- regulation of nuclear materials and radioactive waste safety and security programs
- management of information and information technology
- management of financial programs
- management of corporate functions

2.2 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 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.

2.2.1 Reactor Oversight Process

The regulatory framework for the NRC's Reactor Oversight Process consists 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 and each cornerstone contains performance indicators to ensure that their objectives are being met. The seven cornerstones include: initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and security. Adequate licensee performance, verified through inspection and performance indicator reporting in the seven cornerstone areas, contributes to the NRC's overall reasonable assurance of adequate public health and 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.

2.2.2 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. As of August 2019, 89 of the 97 currently operating units in the United States have had their operating licenses renewed. Based on statements from industry representatives, the Commission expects all but two units to apply for license renewal. As reported in the seventh U.S. National Report, 20 units entered their 41st year of operation (the period of extended operation) between 2014 and 2016. As of August 2019, 6 additional units have entered the period of extended operation as listed below bringing the total to 48 units operating beyond 40 years.

Table 2 Units that Entered the Period of Extended Operation

Year 2017	Year 2018	Year 2019
• Davis-Besse Nuclear Power Station, Unit 1 • Joseph M. Farley Nuclear Plant, Unit 1 • Donald C. Cook Nuclear Plant, Unit 2	• North Anna Power Station, Unit 1 • Edwin I. Hatch Nuclear Plant, Unit 2 • Arkansas Nuclear One, Unit 2	• None

Section 2.3.1.9 of this report discusses subsequent license renewal (i.e., renewal beyond 60 years). Article 14 of this report discusses the license renewal process in detail, including the update to the generic environmental impact statement for license renewal.

2.2.3 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 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 entail major plant modifications. Extended power uprates usually increase power more than 7 percent and require significant modifications to major balance-of-plant equipment. The NRC has approved extended power uprates of up to 20 percent.

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

2.2.4 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, pre-application 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 two 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 increasing 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 three design certification applications, one design certification renewal application, and one early site permit application. All current applications under review 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 Argentina, 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 in which the NRC participates include: (1) cooperation on specific safety design reviews of Westinghouse Electric Company's Advanced Passive 1000 (AP1000), and Korea Electric Power Corporation and Korea Hydro and Nuclear Power Co., Ltd.'s Advanced Power Reactor 1400 (APR1400), and (2) exploration of opportunities to harmonize and converge on regulatory practices in the area of 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.

2.3 Safety and Regulatory Issues, and Regulatory Accomplishments

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

2.3.1 Safety and Regulatory Issues Discussed in the Seventh U.S. National Report

In the seventh U.S. National Report, the NRC staff reported that it was working with the safety and regulatory issues listed in this section. This section presents an update on the following items in alphabetical order:

- baffle-former bolts
- digital instrumentation and control systems
- open phase conditions in electric power system
- Project Aim
- risk-informing regulations and processes
- spent fuel pool (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

2.3.1.1 Baffle-Former Bolts

The core baffle is a portion of the reactor vessel internals in a Westinghouse pressurized-water reactor (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 (through holes in the former plates) in either a downward direction (downflow configuration), or an upward direction (upflow configuration).

Degradation of 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 Information Notice (IN) 98-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants,” dated March 25, 1998. Subsequent inspections in the U.S. plants identified limited degradation at most.

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 because of the pressure differential across the vertical baffle plates. At this time, the most significant degradation of the baffle-former bolts has been observed in Westinghouse four-loop PWR reactors with the downflow configuration and bolts made of type 347 stainless steel. Seven reactors in the United States match this description.

On January 9, 2012, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) issued MRP-227-A, “Pressurized Water 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 shown 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.

From spring 2016 through spring 2017, four U.S. nuclear power plants identified many type 347 stainless steel baffle-former bolts with indications of degradation during ultrasonic inspections following MRP-227-A guidelines. From spring 2017 through spring 2018, several other U.S. nuclear power plants also performed ultrasonic inspections following MRP-227-A guidelines, but identified relatively few stainless steel baffle-former bolts with indications of degradation. In general, these plants replaced potentially degraded bolts and proactively replaced bolts that did not exhibit indications with type 316 stainless steel bolts to improve the design.

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 require the immediate shutdown of any facilities. Between 2016 and 2017, EPRI issued interim guidance to the U.S. nuclear industry on baffle-former bolt inspections in Westinghouse-design PWRs. The NRC’s assessment of this interim guidance is discussed in

"Staff Assessment of EPRI MRP Interim Guidance on Baffle Former Bolts," dated November 20, 2017. Analyses were performed to determine the material condition of the baffle-former bolts that were removed from these plants to provide additional insights on the degradation mechanism. The U.S. nuclear industry formed a working group to consider potential changes to the MRP-227-A inspection regime and also plans to incorporate the interim guidance in future revisions to MRP-227.

On May 21-22, 2019, the NRC met with representatives from the nuclear industry for an exchange of technical information on the industry's materials programs. During this public meeting, EPRI provided an update on baffle-former bolting inspection results in the United States. In all instances except one, the inspection results have been as, or better than, expected. The exception occurred in the spring of 2019 at a 4-loop downflow unit. Visual inspections identified significant degradation of baffle-former bolts. Based on these results, the unit performed ultrasonic testing inspections of all original baffle former bolts. The ultrasonic testing inspections identified a significant number of baffle-former bolts with indications of degradation and a significant clustering of degraded bolts. At the time of the public meeting, the utility was planning the replacement of approximately 272 baffle-former bolts and was considering other potential future actions in fall 2020. Given the uniqueness of this event, the low safety significance, and the licensee's response, the NRC will continue to monitor the industry response to this issue and will take appropriate regulatory actions, if needed.

2.3.1.2 Digital Instrumentation and Control Systems

The NRC staff is implementing a Commission-approved integrated action plan to modernize the NRC's digital instrumentation and control regulatory infrastructure and provide for consistent, predictable, and efficient implementation of digital technology. In parallel, the NRC staff continues to review license amendments for specific digital instrumentation and control systems and to evaluate new reactor applications that fully incorporate highly integrated digital technologies while maintaining adequate protection of the public health and safety. Section 2.3.2.5 of this report provides additional details on the NRC staff efforts in the area of digital instrumentation and control systems.

2.3.1.3 Open-Phase Conditions

Through operating experience, the NRC and its licensees have identified design vulnerabilities associated with open phase conditions in offsite power systems at operating nuclear plants. This condition can degrade the performance capabilities of both offsite and onsite power systems. The licensees have completed modifications related to open phase conditions in 80 percent of the nuclear power plants and modifications for the remaining 20 percent of nuclear power plants are expected to be completed by December 31, 2019. Section 2.3.2.6 of this report provides additional information on the regulatory activities associated with open-phase conditions.

2.3.1.4 Project Aim

In June 2014, the NRC began Project Aim, a strategic initiative to enhance the agency's ability to plan and execute its mission more effectively and efficiently. In June 2015, the Commission approved 19 Project Aim tasks intended to improve efficiency and agility, as well as to right-size the agency, while retaining employees with the appropriate skills needed to perform the NRC's regulatory functions. The staff provided the last of the key deliverables for the Project Aim tasks to the Commission in 2017, with a realized savings of approximately $48 million. In keeping with the NRC's Principles of Good Regulation (independence, clarity, openness, reliability, and efficiency) the NRC is continuing the efforts that began under Project Aim to identify new initiatives and the most effective ways to carry out our safety and security mission. Examples include the merger of the Office of Nuclear Reactor Regulation and the Office of New Reactors, which is expected to be complete by 2020; the enhanced Strategic Workforce Planning (SWP) effort; the improvements in the regional offices' budget formulation and execution; and the Mission Support Standardization and Centralization effort. Through these initiatives, the NRC works to improve the estimates of number of employees and to consider the optimum organizational structure of the agency. Project Aim and related activities facilitated the agency to decrease its size by more than 800 full time equivalent employees since 2011, while sustaining the commitment to its safety and security mission. Future agency initiatives will be reported under the Innovation and Transformation efforts. Section 2.3.2.10 of the report provides additional details.

2.3.1.5 Risk-Informing Regulations and Processes

While the NRC has had a long standing approach to risk-informed performance-based regulations and inspection oversight of reactors, the NRC is advancing the use of risk-information in its decisionmaking processes through the Risk-Informed Decisionmaking (RIDM) effort. As part of this effort, the agency is sensitive to the need to remain consistent with the defense-in-depth philosophy, maintain sufficient safety margins, and ensure appropriate performance monitoring. Significant activities include completing an assessment of challenges to further progress in RIDM in licensing actions for operating reactors, and implementing an action plan to communicate strategies and guide efforts to improve RIDM. To broaden the effort across the agency, this high priority initiative receives support from management at all levels across the agency, is guided by the NRC's Risk Informed Steering Committee (RISC), and incorporates stakeholder input through frequent public meeting opportunities. Section 2.3.2.8 of this report provides additional details.

2.3.1.6 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," in Appendix A to 10 CFR Part 50; and other equivalent regulatory criteria. The NRC includes a similar requirement 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. Neutron absorbing materials are expected to be used for the life of the pools for the existing fleet and all new construction. As previously reported, the NRC has taken several actions to address the potential degradation of these materials.

For example, the NRC is performing confirmatory research on SFP neutron-absorbing materials used in the U.S. commercial nuclear power industry, including the surveillance methodologies and intervals used by the industry. The NRC is also following research being conducted by EPRI.

NRC Generic Letter (GL) 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools," dated April 7, 2016, asks licensees to provide information demonstrating 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 comply with the licensing and design basis, and applicable regulatory requirements. The NRC has completed its review of responses to the GL and issued a closeout letter to each licensee. "Completion of Review of Power Reactor Licensee Responses to Generic Letter 2016-01, Monitoring of Neutron Absorbing Materials in Spent Fuel Pools," dated November 27, 2018, presents an overview of the NRC's review.

2.3.1.7 Staff Readiness to Transition Plants from Construction to Operations

With respect to the NRC's readiness to transition new reactors from construction to operations, the agency established a staff-level transition working group in 2013 to develop an integrated plan that identified all regulatory functions necessary to support the transition of the 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 NRC staff is currently tracking and reporting to management on the resolution of these recommendations on a regular basis.

In response to one of the readiness issues, the NRC published the "Implementation Plan To Ensure NRC Staff Readiness for AP1000 Operations," on November 16, 2017. The plan offers the vision and strategy for how the NRC will prepare for AP1000 operations, specifically applicable to Vogtle Electric Generating Station, Units 3 and 4. The NRC continues to plan for the transition from construction to operations, including changes to existing oversight programs to incorporate the AP1000 design.

In March 2018, the NRC established the Vogtle Readiness Group to provide oversight and management direction to ensure that the Vogtle units under construction meet regulatory requirements and are safe to operate. The Vogtle Readiness Group serves as the focal point for project status and for coordination through commercial operation of the Vogtle Electric Generating Station, Units 3 and 4, and serves as the hub for communications with the Commission, the Executive Director for Operations and the directly reporting offices, the licensee, and other external stakeholders.

The Vogtle Readiness Group issued the "Charter for Instituting the Vogtle Readiness Group to Oversee the Vogtle Units 3 and 4 Transition to Operations," dated March 12, 2018, and has developed an integrated project plan, based on the licensee's schedule, to ensure that the NRC is prepared to complete the activities within its control (e.g., licensing; inspections; and inspection(s), test(s), analysis (analyses) and acceptance criterion/criteria (ITAAC) closure). The integrated project plan lists the regulatory milestones for the transition of Vogtle's construction to operations and it is used to identify potential critical path areas. These areas include initial testing, implementation of operational programs, transition to operations, cybersecurity, emergency preparedness, security transition (nonsafeguards), transition from construction to operating reactor oversight, and key communications with stakeholders. It also includes the development of inspection and licensing support documents.

2.3.1.8 Staff Readiness to Transition Plants from Operation to Decommissioning

When a licensee 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.

Several NRC regulations set out the requirements for decommissioning a nuclear power plant.[2] 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.

However, most regulations in 10 CFR Part 50, as well as other parts of the NRC's regulations were intended to apply 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 shutdown reactor are significantly lower than the risks from operating reactors, certain reactor regulations and specific license conditions may no longer be necessary for 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 (Kewaunee Power Station; Crystal River Nuclear Generating Plant, Unit 3; San Onofre Nuclear Generating Station, Units 2 and 3; and Vermont Yankee Nuclear Power Station). These were the first reactors to permanently cease operations since 1998. In 2016, Fort Calhoun Station permanently ceased operation, in 2018, Oyster Creek Nuclear Generating Station permanently ceased operation, and in 2019, Pilgrim Nuclear Power Station permanently ceased operation. Licensees for 11 other reactor units have announced an intent to permanently cease operations by 2025: Three Mile Island Nuclear Station, Unit 1; Davis-Besse Nuclear Power Station; Duane Arnold Energy Center; Perry Nuclear Power Plant; Beaver Valley Power Station, Units 1 and 2; Indian Point Nuclear Generating, Units 2 and 3; Palisades Nuclear Power Station; and Diablo Canyon Power Plant, Units 1 and 2. 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 certain license conditions and technical specifications. The NRC staff reviews each request with a focus on safety and the individual circumstances at each site and decides 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.

Following the terrorist attacks of September 11, 2001, and the 2011 Fukushima Dai-ichi nuclear accident in Japan, there have been revised regulatory requirements for operating plants related to emergency preparedness and security.

2 See 10 CFR Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination," and 10 CFR 50.75, 50.82, 51.53, and 51.95.

In order to consider the applicability of these revised requirements for future plant decommissioning, the Commission directed the staff to develop a proposed decommissioning rule[3] to address issues such as:

- the graded approach to emergency preparedness
- lessons learned from the plants that have already gone or are currently going through the decommissioning process
- the advisability of requiring a licensee's Post-Shutdown Decommissioning Activities 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 nongovernmental 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, which is discussed in more detail in Section 2.3.3.3 of this report.

2.3.1.9 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."

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. SRM-SECY-14-0016 identified four technical issues related to subsequent license renewal for further consideration: reactor pressure vessel neutron embrittlement at high fluence, irradiation assisted stress corrosion cracking of reactor internals and primary system components, concrete and containment degradation, and electrical cable qualification and condition assessment.

To address the four issues and to effectively leverage resources and knowledge, the NRC is collaborating on research activities with both domestic and international partners.

Licensees and applicants provide the technical basis to support their safety analysis and application for subsequent license renewal. The NRC staff then conducts confirmatory research to independently verify licensee data, determine safety margins, and explore uncertainties. The

3 SRM-SECY-14-0118, "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," dated December 30, 2014.

NRC continues to track industry's work on aging management, evaluate areas for research, gather data to assess the effectiveness of licensee's aging management programs, and provide confirmatory research on the results of industry's work. Results from the 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.

As of August 2019, 89 of the 97 currently operating nuclear reactors have received license extensions to 60 years. The NRC is currently reviewing subsequent license renewal applications for Turkey Point Nuclear Generating, Units 3 and 4, Peach Bottom Atomic Power Station, Units 2 and 3, and Surry Power Station, Units 1 and 2. On November 7, 2017, the Commission received a formal letter of intent to pursue a subsequent license renewal for North Anna Power Station, Units 1 and 2, in 2020. In its review of subsequent license renewal applications, the NRC staff developed guidance documents to address the unique aging management needs for a subsequent license renewal. In July 2017, the NRC published NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," and NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants." In December 2017, the NRC staff published NUREG-2221, "Technical Bases for Changes in the Subsequent License Renewal Guidance Documents NUREG-2191 and NUREG-2192," and NUREG-2222, "Disposition of Public Comments on the Draft Subsequent License Renewal Guidance Documents NUREG-2191 and NUREG-2192." The staff developed these guidance documents by making the necessary revisions to the existing license renewal guidance documents for 40 years of operation (e.g., 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 December 2010) to account for expected aging management needs for the 60 to 80 year operating period. Sections 2.3.2.9, 2.3.3.1, and 14.1.4.3 of this report describe the subsequent license renewal activities in more detail.

2.3.2 Current Safety and Regulatory Issues

The NRC and its licensees are evaluating and resolving the following potential safety and regulatory issues. These items are presented in alphabetical order.

- accident tolerant fuel
- assessment of debris accumulation on sump performance
- changes to the Reactor Oversight Process
- clarifying the backfit process
- digital instrumentation and control systems
- open-phase conditions
- proposed rulemaking on emergency preparedness for small modular reactors and other new technologies

- risk-informed decisionmaking (RIDM)
- subsequent license renewal challenges
- transformation at the NRC

2.3.2.1 Accident Tolerant Fuel

The NRC performs fuel system safety reviews to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage does not prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained.

The U.S. nuclear industry, with the assistance of the U.S. Department of Energy (DOE), is seeking to develop and deploy new fuel technologies that are expected to perform better than currently licensed fuels during normal operation and under transient and accident conditions. Near-term accident tolerant fuel designs, which the industry is pursuing for deployment by the mid-2020s, will have relatively small departures from today's nuclear fuel designs. These departures include specially designed additives to standard fuel pellets; coatings applied to the outside diameter of standard claddings; and ferritic steel claddings intended to reduce corrosion, increase wear resistance, and reduce the production of hydrogen under accident conditions.

To increase regulatory stability and certainty, along with enhancing and optimizing NRC review, the staff has developed a "Project Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuels," dated September 25, 2018, which includes a new paradigm for fuel licensing. The plan addresses the complete fuel cycle, including fuel fabrication, fresh fuel transport, in-reactor requirements, and spent fuel storage and transportation and outlines the NRC's strategy for enhancing our regulatory infrastructure to support thorough and timely licensing reviews of accident tolerant fuel designs. The staff believes that adherence to this strategy, which significantly enhances engagements with the nuclear fuel vendors early in the research and development phase, will benefit all the agency's stakeholders through the planned deployment of accident tolerant fuel designs.

2.3.2.2 Assessment of Debris Accumulation on Sump Performance

Generic Safety Issue (GSI)-191 evaluated the possibility that, after a loss of coolant accident (LOCA) in a PWR, debris accumulating on the emergency core cooling system sump strainer may result in degradation of the system. In order to address this possibility, all PWR licensees made physical and operational improvements to their plants. GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004, requested licensees to document actions and evaluations to determine the adequacy of these changes and to address technical issues related to debris that may pass through the strainers and cause in-vessel issues.

SECY-12-0093, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated July 9, 2012, proposed three options for licensees to choose from to close GSI-191. The Commission approved these options on December 14, 2012. Using these options, 21 PWR units resolved the issue by 2017.

The remaining 44 units indicated that they intend to respond to GL 2004-02 using deterministic or risk-informed evaluations for the strainer and in-vessel issues.

In-vessel issues, which were not originally part of GSI-191, required extensive testing and evaluation. To address these issues on a plant-specific basis, the PWR Owners Group submitted a topical report, WCAP-17788, “Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090),” dated July 17, 2015. The staff concluded that in-vessel issues are generically of low safety significance; therefore, this is no longer under review. The staff’s analysis is documented in “Technical Evaluation Report of In-Vessel Debris Effect,” dated June 13, 2019.

For BWRs, the NRC and the nuclear industry conducted research and testing from 1992 to 2001 to resolve the issue of debris blockage of sump strainers. During that time, the staff issued Bulletin 95-02, “Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode,” on October 17, 1995, and Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” on May 6, 1996. Both bulletins dealt with ensuring that debris generated during a LOCA would not clog emergency core cooling system suction strainers. After testing, analysis, and plant modifications, which included upgraded strainers, the NRC concluded that all BWR licensees had sufficiently responded to the requested actions. The staff documented this conclusion in the “Completion of Staff Reviews of NRC Bulletin 96-03” memorandum dated October 18, 2001.

Following the resolution of the issue for BWRs in 2001, the work on similar issues for PWRs described above resulted in updated knowledge of the phenomena associated with strainer and downstream blockage issues. On April 10, 2008, the NRC issued a letter, “Potential Issues Related To Emergency Core Cooling Systems (ECCS) Strainer Performance at Boiling Water Reactors,” encouraging BWR Owners Group members to develop a comprehensive plan to address the issues based on the updated knowledge. In this letter, the NRC identified emergency core cooling system issues related to post-LOCA debris that should be evaluated to ensure that the earlier BWR resolution was still conservative.

The BWR Owners Group conducted its analysis under a voluntary initiative and adopted a risk-informed approach to address the identified issues. The group used Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 2, dated May 2011, to provide a consistent and logical framework to assess the risk significance of the identified issues. Although the BWR Owners Group evaluation was not a license amendment request, the NRC staff evaluated it and concluded that the group had adequately addressed each of the five principles of RIDM in RG 1.174. The staff also concluded that the effects of the identified issues would have low risk significance for emergency core cooling system performance. The NRC notified the BWR Owners Group of its findings in letter, “Closure of Potential Issues Related to Emergency Core Cooling Systems Strainer Performance at Boiling Water Reactors,” dated June 29, 2018. The NRC considers the issue to be closed for BWRs.

2.3.2.3 Changes to the Reactor Oversight Process

Many stakeholders—including the industry, public, and international community—recognize the Reactor Oversight Process as an effective oversight program that ensures safety. The Reactor Oversight Process continuously evolves based on the NRC’s self-assessments, lessons learned activities, and feedback from internal and external stakeholders.

For example, the NRC has recently established a new format for inspection reports to make them more user friendly and is working on a software application that will allow NRC inspectors to automatically generate online inspection reports to increase efficiency. The software is expected to decrease the time it takes for NRC inspectors to create inspection reports, increase their time inspecting in the field, and increase consistency among inspection reports.

The staff continues to work on a project to improve the process for screening inspection findings of minor safety or security significance (green findings), which make up most of the issues the NRC finds during inspections. The staff often spends much time determining whether an issue is minor and recognizes the benefits of improving inspector guidance in that area.

The NRC has worked with stakeholders to develop streamlined processes and procedures for inspecting engineering programs. The inspection will focus on operating experience, aging management, and changes to the design basis and probabilistic risk assessment (PRA) models. The NRC staff proposed modifying engineering-related inspection procedures to incorporate these topics and reduce the frequency of inspections (e.g., quadrennial instead of triennial). This proposal is described in SECY-18-0113, “Recommendations for Modifying the Reactor Oversight Process Engineering Inspections,” dated November 13, 2018. The Commission is currently considering the staff’s recommendations for these inspections.

The NRC staff also recommended changes to the treatment of safety significant inspection findings and performance indicators to provide an incentive for licensees to prepare for supplemental inspections in a timely manner commensurate with their safety significance. The NRC staff completed a comprehensive review of all reactor safety inspection procedures, suggesting changes in some inspection sample sizes, and identifying risk-informed changes to the significance determination process for emergency preparedness. These changes will improve efficiency and allow inspectors to focus efforts on safety significant issues while still ensuring the safety cornerstone objectives continue to be met.

The NRC is also conducting a comprehensive review of the problem identification and resolution inspections, an effectiveness review of the changes made to the cross-cutting issues process, and potential changes to inspections for radiation protection and independent spent fuel storage installations, among others. This proposal is described in SECY-19-0067, “Recommendations for Enhancing the Reactor Oversight Process,” dated June 28, 2019. The Commission is reviewing the staff’s recommendations for this proposal.

2.3.2.4 Clarifying the Backfit Process

Backfitting is “the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission’s regulations or the imposition of a regulatory staff position interpreting the Commission’s regulations that is either new or different from a previously applicable staff position.” The imposition of a backfit is done only after formal, systematic review to ensure that the resulting changes are properly justified and suitably defined. The requirements for properly justifying backfitting actions for nuclear power reactors are found in 10 CFR 50.109, “Backfitting.” The NRC’s “issue finality” requirements in 10 CFR Part 52 are intended to accomplish the same objective as the 10 CFR 50.109 requirements for new reactor applicants by limiting changes to prior Commission decisions made on design and site suitability as part of the 10 CFR Part 52 licensing process.

The NRC is updating and improving the implementation of its backfitting and issue finality requirements. In 2015 and 2016, the NRC received feedback from external stakeholders regarding a potential lack of rigor in NRC adherence to the regulatory framework of its backfitting process. As a result of this feedback, the NRC Executive Director for Operations tasked the Committee to Review Generic Requirements (CRGR) and the offices involved in backfitting to reevaluate the NRC guidance, training, and knowledge management in this area. Also in 2016, the Commission directed the staff to revise the guidance on backfitting to reflect the Commission's adoption of recent guidance from the NRC's General Counsel that interpreted a recent decision from the United States Supreme Court. In June 2017, the CRGR issued its evaluation of the agency's backfitting and issue finality program, including recommended enhancements.

The enhancements were principally aimed at implementing a more rigorous and disciplined backfitting process and centered on (1) updating and improving the backfitting and issue finality guidance, primarily Management Directive (MD) 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013, and NUREG-1409, "Backfitting Guidelines," dated July 1990, (2) conducting near-term and follow-on backfitting training for NRC staff and managers, and (3) capturing, storing, and transferring backfitting knowledge through the use of knowledge management and knowledge transfer tools.

As of July 2019, the NRC has made significant progress in this ongoing effort including the following:

- Issued an internal NRC Yellow Announcement (YA)-17-0077, "Commission Direction and Staff Actions Related to Backfitting and Issue Finality," in August 2017, to more broadly communicate to the NRC staff the Commission direction on backfitting guidance

- The Commission issued SRM-SECY-18-0049, "Management Directive and Handbook 8.4, Management of Backfitting, Issue Finality, and Information Collection" on May 29, 2019, reflecting changes in the title of the document and the approval of the backfitting and forward fitting guidance

- Conducted backfitting training for NRC staff from November 2017 through January 2018, and in-depth workshops in June and July 2018. More than 1,400 staff members received the training materials, which have been made public.

- Revised the CRGR Charter

- Created an internal SharePoint site to share backfitting knowledge, experience, and lessons learned

- Created a backfitting community of practice to provide additional resources and enhance consistency in the application of the backfitting provisions across the agency. Backfitting subject matter experts in every region and office have been identified.

The NRC staff is currently working on issuing the revised MD 8.4, as approved by the Commission. In the long-term, the NRC plans to build backfitting training into the existing agency training infrastructure to ensure continued training that reflects the various roles and responsibilities of staff members. Section 14.1.5.2 of this report provides additional information about the NRC's backfitting process.

2.3.2.5 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. However, the efficiency and predictability of the processes can be improved. In May 2016, the staff submitted to the Commission for review and approval an integrated 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. The Commission approved the staff's planned approach in October 2016.

A Steering Committee, consisting of NRC senior managers, provides oversight for updating and implementing the integrated action plan to ensure a sound strategy to modernize the NRC's digital instrumentation and control regulatory infrastructure. The plan encompasses all NRC digital instrumentation and control activities and regulatory challenges. Examples include incorporation of Institute of Electrical and Electronics Engineers (IEEE) standards into NRC regulations, updates to the NRC's policy on common-cause failure, and guidance for applicants and licensees implementing digital instrumentation and control systems. The plan ensures 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 advances in nuclear applications of digital technology. Also, the integrated action plan ensures 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.

The staff continues to make progress in implementing the Commission-approved integrated action plan. For example, the staff improved guidance to clarify the regulatory process that licensees use to make digital modifications without prior NRC approval as described in 10 CFR 50.59, "Changes, Tests and Experiments." The staff has received positive feedback and licensees are implementing digital modifications using this guidance. The staff continues to make progress in streamlining the licensing review process for major digital upgrades by issuing interim staff guidance that would reduce the scope of licensee document submittals and would provide an alternative for earlier approval. The staff has also reassessed its position on addressing common cause failure and will clarify regulatory guidance for addressing common cause failure and associated defense-in-depth analysis with a graded approach.

In addition, the staff is focusing on high priority activities with the greatest near-term tactical impact. The long-term strategy is to evaluate and strategically implement broader improvements in the NRC's digital instrumentation and control regulatory infrastructure. This includes the potential review and acceptance of International Electrotechnical Commission standards for digital systems, as an alternative to domestic U.S. standards (i.e., IEEE standards). The staff continues to engage with industry in public meetings to discuss challenges and priorities that may be unique to specific digital instrumentation and control stakeholders.

As these modernization activities are ongoing, the NRC staff continues to review and approve license amendments for specific digital instrumentation and control systems. The NRC is also evaluating new reactor applications that fully incorporate highly integrated digital technologies while maintaining adequate protection of the health and safety of the public.

Section 18.3.2.2 of this report provides additional information about the staff's technical review of digital instrumentation and control system.

2.3.2.6 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 domestically and internationally. An open phase condition may occur because of various faults such as circuit breaker poles not opening or closing, or the failure of transformer bushings or line insulators that leads to a loss of circuit continuity. This type of fault creates voltage and current imbalances in electrical power systems that may be detrimental to operating equipment. An open phase condition, if not detected and disconnected in a timely manner, may lead to the degrading or tripping of redundant equipment, which could compromise the safe shutdown capability of the plant.

On January 30, 2012, an operating event at Byron Station, Unit 2,[4] revealed a significant design vulnerability, which resulted in the loss of safety functions for electric power systems. The unit's 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 systems, structures, and components (SSCs) important to safety. The NRC 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 operating experience identified similar design vulnerabilities at South Texas Project, Unit 2; Beaver Valley Power Station, Unit 1; Nine Mile Point Nuclear Station, Unit 1, and James A. FitzPatrick Nuclear Power Plant.[5] In addition, operating experience has identified three similar international events at reactors located in Canada, Sweden, and the United Kingdom.

The electric power system design at the majority of U.S. nuclear power plants did not include provisions to minimize the probability of losing electric power from any of the remaining supplies resulting from, or coincident with, the loss of power from the transmission network caused by an open phase condition. Therefore, on July 27, 2012, the NRC staff issued Bulletin 2012-01, "Design Vulnerability in Electric Power System." The NRC staff reviewed the information that the licensees provided and concluded that this design vulnerability exists at all operating plants, except for Seabrook Station, because of plant-specific switchyard features.

In response to this operating experience, most licensees have chosen to implement the Nuclear Energy Institute (NEI)'s voluntary industry initiative, which is discussed in letters dated October 9, 2013; March 16, 2015; and September 20, 2018. The objective of the voluntary industry initiative is to ensure that important-to-safety functions remain available in the event of an open phase condition. The voluntary industry initiative also addresses the installation of plant

4 Byron Station, Units 1 and 2, LER 2012-001-00, dated March 30, 2012, and LER 2012-001-01, dated September 8, 2012.

5 South Texas Project, Unit 2, Licensee Event Report (LER) 2001-001, dated April 3, 2001; Beaver Valley Power Station, Unit 1, LER 2007-002-00, dated January 25, 2008; Nine Mile Point Nuclear Station, Unit 1, LER 2005-04, dated February 17, 2006; and James A. FitzPatrick Nuclear Power Plant LER 2005-006, dated February 13, 2006.

modifications (open phase isolation system) that allow plant operators to identify compensatory actions needed to detect and isolate offsite power sources due to open phase conditions. The industry has completed modifications related to open phase conditions in approximately 80 percent of the nuclear power plants and the remaining modifications are expected to be completed by December 31, 2019.

NRC regional inspectors used Temporary Instruction 2515-194, "Inspection of the Licensees' Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems (NRC Bulletin 2012 01)," dated October 31, 2017, to verify the implementation of the voluntary industry initiative. In June 2018, the NRC conducted pilot inspections at four nuclear plants, representing four fundamental open phase isolation system designs.

In February 2019, NEI provided to the NRC a draft paper "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights," describing a risk-informed approach to address open phase isolation system protective action requirements. Specifically, the document compares the risk when operating with automatic functions to isolate a power supply affected by an open phase condition and the risk when operating with manual actions. NEI has revised the voluntary industry initiative to allow, as an option, the use of this risk-informed approach. The NRC staff received the revised voluntary industry initiative and the revised risk guidance document in June 2019. The NRC staff is preparing to audit the implementation of the risk option at the two pilot sites identified by the industry. Based on the results of the audits, the NRC staff will determine what inspections at the nuclear power plants that choose to implement this revised option are warranted. The NRC anticipates using the information from the voluntary industry initiative, the results of the initial inspections, and the staff evaluation of the adequacy of the voluntary industry initiative to close Bulletin 2012-01.

2.3.2.7 Proposed Rulemaking on Emergency Preparedness for Small Modular Reactors and Other New Technologies

Current emergency preparedness requirements and guidance, initially developed for large light-water reactors and nonpower reactors, do not consider small modular reactors, non-light water, and other new technologies, such as medical isotope production facilities. Consistent with Commission direction in SRM-SECY-16-0069, "Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies," dated June 22, 2016, the NRC is examining these issues in the rulemaking process, recommending revisions to the regulations, and developing implementation guidance documents. The NRC staff's proposed emergency preparedness requirements and implementing guidance would adopt a consequence-oriented, risk-informed, performance-based, and technology-inclusive approach. The staff's objective for this rulemaking is to create a set of requirements that would achieve the following:

(1) Continue to provide reasonable assurance that adequate protective measures can and will be implemented for a small modular reactor or other new technologies.

(2) Promote regulatory stability, predictability, and clarity.

(3) Reduce requests for exemptions from emergency preparedness requirements.

(4) Recognize technology advances embedded in design features.

(5) Credit safety enhancements in evolutionary and passive systems.

(6) Credit smaller reactors' and nonlight-water reactors' potential benefits associated with postulated accidents, including slower transient response times, and relatively small and slow release of fission products.

This rule and guidance could affect existing and future facilities to be licensed after the effective date of the final rule. These applicants and licensees would have the option to develop a performance-based emergency preparedness program, rather than using the existing deterministic emergency preparedness requirements in 10 CFR Part 50.

On April 13, 2017, the NRC issued the draft regulatory basis, "Emergency Preparedness for Small Modular Reactors and Other New Technologies," and the associated regulatory analysis in Volume 82 of the *Federal Register* (82 FR 17768) and held a public meeting to discuss the documents. The NRC considered all public comments and published the final regulatory basis on November 15, 2017 (82 FR 52862). The NRC staff used the regulatory basis and the associated regulatory analysis to inform the development of the draft proposed rule. On October 12, 2018, the staff submitted SECY-18-0103, "Proposed Rule: Emergency Preparedness for Small Modular Reactors and Other New Technologies," to the Commission for review and approval.

This draft proposed rule includes the following major provisions:

- a new alternative performance-based emergency preparedness framework, including requirements for demonstrating effective response in drills and exercises for emergency and accident conditions

- a hazard analysis of any NRC-licensed or nonlicensed facility located contiguous to small modular reactor or other new technology facility that considers any hazard that would adversely affect the implementation of emergency plans

- a scalable approach for determining the size of the plume exposure pathway emergency planning zone

- a requirement to describe ingestion response planning in the emergency plan, including the capabilities and resources available to prevent contaminated food and water from entering the ingestion pathway

The Commission is currently evaluating the staff's recommendations in SECY-18-0103.

2.3.2.8 Risk-Informed Decisionmaking

The NRC is advancing the use of risk information in its decisionmaking processes. As part of this effort, the NRC is sensitive to the need to be consistent with the defense-in-depth philosophy, maintain sufficient safety margins, and ensure appropriate performance monitoring. Activities in this area occur under the RIDM effort. Significant activities include an assessment of challenges to further progress in RIDM and implementation of an action plan to communicate strategies and guide efforts to improve RIDM. These initiatives will reinforce the expectation to use risk insights holistically at the early stages of regulatory activities to more efficiently guide the agency's efforts, improve communications, and achieve consistency to advance the application of RIDM beyond the nuclear reactor safety program.

The NRC's vision is that RIDM—and notably, the resulting safety focus and efficiency benefit—will be applied broadly across all regulatory activities.

<u>NRC's Risk Informed Steering Committee</u>. The RISC is an NRC senior management committee that provides strategic direction to the NRC staff to advance the use of RIDM in licensing, oversight, rulemaking, and other regulatory areas, consistent with the Commission's policy statement "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," dated August 16, 1995, in the *Federal Register* (60 FR 42622). 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 Material Safety and Safeguards, Nuclear Security and Incident Response, and Nuclear Reactor Regulation, as well as a Regional Administrator or a Deputy Regional Administrator.

The nuclear power industry has its own RISC, which is a counterpart to the NRC's committee. Membership of the industry's RISC consists of licensee chief nuclear officers and other senior level executives, as well as representation from NEI.

<u>Action Plan To Further RIDM and Address Challenges</u>. The Commission directed the staff to develop plans for increasing staff capabilities to use RIDM in regulatory activities. In SECY-17-0112, "Plans for Increasing Staff Capabilities To Use Risk Information in Decision-Making Activities," dated November 13, 2017, the staff communicated several challenges associated with advancing RIDM and provided strategies to address them. Some challenges stem from the NRC staff having varying degrees of awareness and knowledge of RIDM processes and applications. Others include the staff not having fully integrated reviews to include complementary insights from traditional engineering and risk assessment approaches and a lack of guidance for using risk insights in licensing actions not formally submitted using the NRC framework for risk-informed licensing actions. In SECY-17-0112, the staff also discussed a multifaceted approach to overcoming these challenges.

The NRC implemented an action plan to enhance the integration of risk information into the agency's decisionmaking practices and processes to improve the technical basis for regulatory activities, increase efficiency and improve effectiveness. The comprehensive plan is focused on operating reactors licensing and has two phases as well as communication strategies throughout the entire action plan. Phase I focused on collecting data, evaluating, and analyzing RIDM-related tasks to generate findings and recommendations. Phase II focused on implementation of Phase I recommendations through 13 action items, including revising agency guidance documents and training staff.

<u>RIDM Knowledge Management Efforts</u>. The knowledge management effort for RIDM seeks to broaden the understanding of risk beyond a quantitative value to one that considers quantitative risk as one of five key principles along with defense-in-depth philosophy, safety margins, performance measurement strategies and regulatory compliance. One key accomplishment in this area is the completion of a new pilot course for managers that provides perspectives on how risk and deterministic information are used together to make regulatory decisions, to review risk-informed licensing guidance and recent actions, and to illustrate risk management tools and practices at utilities. The NRC will use this pilot training to inform development of future training options for managers.

Improving Use of Risk Information in Licensing Actions. The RIDM action plan recognizes that improvements to the licensing processes should be specific to how risk information is used in reviewing a particular type of licensing action. In general, the NRC plans to enhance staff knowledge and expand on an already existing framework for a graded approach to these types of licensing reviews. The existing framework is documented in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

For formal risk-informed licensing actions, the NRC has sought to optimize the review process. For example, integrated review teams consisting of several technical disciplines review licensing actions; management promptly communicates lessons learned; staff practices discipline when requesting additional information from the licensees; and the NRC enlists the support of independent third-party reviewers to address a surge in these types of licensing actions. Finally, the RISC provides additional management oversight for formal risk-informed licensing actions.

U.S. nuclear utilities are actively pursuing efforts to adopt RIDM tools within their licensing basis to gain operational and engineering flexibilities. Two specific areas where RIDM tools are currently being applied are in the review of licensing actions to implement 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," and in the review of licensing actions to modify certain technical specifications based on risk information.

In 10 CFR 50.69, the NRC provides requirements for implementing a process to risk-inform the characterization of SSCs for the purpose of tailoring applicable regulatory requirements (e.g., inservice testing, quality assurance measures). A risk-informed categorization process allows the NRC to focus regulatory attention on issues that have the greatest potential to impact public health and safety and focuses licensee attention on the most risk significant equipment. Broad implementation of 10 CFR 50.69 will represent a significant step toward realization of the benefits of RIDM. Licensees have expressed interest in applying this rule and submitting numerous licensing applications in the near future. To support this approach, the nuclear industry has developed a template for applications of this formal risk-informed licensing action and the NRC has endorsed an equipment categorization process through RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Revision 1, dated May 2006. Additionally, the NRC has third party reviewers under contract to assist in reviewing these licensing actions.

The NRC has reviewed licensing applications involving the establishment of a risk management approach for certain limiting conditions for operation contained within technical specifications under Technical Specification Task Force (TSTF)-425, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," Revision 3, dated March 18, 2009, and TSTF-505, "Provide Risk-Informed Extended Completion Times (CTs) – RITSTF Initiative 4b," Revision 2, dated November 21, 2018. TSTF-425 provides a risk-informed methodology to identify, assess, implement, and monitor proposed changes to surveillance requirement frequencies in technical specifications. TSTF-505 allows licensees to modify selected required actions to permit extended completion times, if risk is assessed and managed within an acceptable configuration risk management program. The NRC staff continues to work on these initiatives to add a risk-informed component to the standard technical specifications. These initiatives are intended to maintain and improve safety by incorporating risk assessment and management techniques in the technical specifications, while reducing unnecessary burden.

2.3.2.9 Subsequent License Renewal

As discussed in Section 2.3.1.9 of this report, the NRC staff is currently reviewing subsequent license renewal applications for Turkey Point Nuclear Generating, Units 3 and 4; Peach Bottom Atomic Power Station, Units 2 and 3; and Surry Power Station, Units 1 and 2. Using lessons learned from reviewing initial license renewal applications, the NRC staff aims to complete the safety and environmental reviews for subsequent license renewal applications within 18 months of accepting the application. Any person whose interest may be affected by the issuance of the renewed license can request a hearing or a petition to intervene in accordance with 10 CFR 2.309, “Hearing Requests, Petitions to Intervene, Requirements for Standing, and Contentions.”

The target review timeline of 18 months assumes the licensee submits a high-quality application and responds promptly and completely to the NRC’s requests for additional information. Hearings and petitions to intervene could also affect the staff’s schedule for issuing a decision. Consistent with SRM-SECY-14-0016, the staff will keep the Commission informed on the following technical issues related to subsequent license renewal: reactor pressure vessel neutron embrittlement at high fluence, irradiation assisted stress corrosion cracking of reactor internals and primary system components, concrete and containment degradation, and electrical cable qualification and condition assessment.

On January 31, 2018, the NRC received the subsequent license renewal application for Turkey Point, Units 3 and 4. The NRC accepted the application for review in April 2018. As of July 2019, the NRC staff’s review is ongoing. The NRC staff has encountered challenges in the review associated with aging management of irradiated concrete structures, irradiation of reactor vessel support steel, and plant-specific analyses of topical reports referenced in the application that the staff has not previously reviewed and approved. The NRC staff issued the draft Supplemental Environmental Impact Statement on March 29, 2019 and the safety evaluation report with open items on July 22, 2019. Three petitions to intervene and requests for hearing were filed. On March 7, 2019, the Atomic Safety and Licensing Board denied one petition; granted two petitions; admitted four environmental contentions for litigation; and referred to the Commission one portion of its ruling concerning the applicability of the environmental regulations to subsequent license renewal applications. On April 9, 2019, one of the admitted petitioners filed a notice of its withdrawal from the proceeding. On July 8, 2019, the Board dismissed the two remaining contentions. New and amended contentions were filed on June 24, 2019.

On July 10, 2018, the NRC received the subsequent license renewal application for Peach Bottom, Units 2 and 3. The NRC accepted the application for review in August 2018. As of July 2019, the review is ongoing. The NRC staff has encountered challenges in the review associated with plant-specific analyses of topical reports referenced in the application that the staff has not previously reviewed and approved. One petition to intervene and request for hearing was filed. The oral argument on the contention admissibility was held on March 27, 2019. On June 20, 2019, the Board found that, although the petitioner had demonstrated standing to intervene, neither of its two proposed contentions was admissible. Therefore, the Board denied the petition to intervene and request for hearing and terminated the proceeding. On July 15, 2019, the petitioner filed an appeal from the Board’s decision.

Additional information on these petitions to intervene and requests for hearing on the Turkey Point and Peach Bottom applications is located on the NRC’s electronic hearing dockets Web site (https://adams.nrc.gov/ehd) under “Turkey Point 50-250 & 50-251-SLR” and “Peach Bottom 50-277 & 50-278-SLR,” respectively.

On October 16, 2018, the NRC received the subsequent license renewal application for Surry, Units 1 and 2. The NRC accepted the application for review in December 2018. To date, no hearing requests have been received. As of July 2019, the review is ongoing. The NRC staff has encountered challenges in the review associated with the aging management of steel components of the reactor pressure vessel support assembly, the analysis of the impacts on Atlantic sturgeon critical habitat or essential fish habitat, and plant-specific analyses of topical reports referenced in the application that the NRC staff has not previously reviewed and approved.

2.3.2.10 Transformation at the NRC

NRC regulations have served the United States well and will continue to ensure safety and security; however, some of them were written to be technology-specific and do not easily accommodate new technologies, such as advanced reactors and digital instrumentation and control. It is the responsibility of the NRC to ensure that its regulations continue to provide an acceptable level of safety in a way that accommodates new technologies. Further, the NRC wants to ensure that its regulatory framework does not present a barrier to safety enhancements. This is an important step in continuing to be a relevant and modern regulator in the future.

To encourage innovation and provide more focus on transformation, on January 4, 2018, the NRC’s Executive Director for Operations issued a message to all staff on “Innovation and Transformation at the NRC.” That message described the need for the NRC to become more agile, efficient, and effective in how it regulates new and developing technologies such as accident tolerant fuels, new materials, new manufacturing approaches, digital instrumentation and control, and small modular and advanced reactor designs. Subsequently, a team of NRC staff members was given the task of identifying potential transformative changes to the NRC’s regulatory framework, culture, and infrastructure. The NRC’s transformation team gathered information on potentially transformative approaches by interacting with internal and external stakeholders. These interactions included internal discussions at regional, office, and division-level meetings as well as meetings with nuclear industry representatives, other Federal agencies, international counterparts, and nongovernmental organizations. The transformation team identified potential transformational changes to the NRC’s regulatory framework, culture, and infrastructure to enhance the agency’s effectiveness, efficiency, and agility in regulating new and novel technologies.

The transformation team submitted SECY-18-0060, “Achieving Modern Risk-Informed Regulation,” to the Commission on May 23, 2018, recommending transformative changes in four areas:

(1) transform the agency licensing review process by developing an agencywide process and organizational tools that expand the systematic use of qualitative and quantitative risk and safety insights, thereby, enabling the staff to scale the scope of review and level of detail needed to make a finding of reasonable assurance of adequate protection

(2) revise 10 CFR 50.59 and other similar requirements to allow additional flexibility for licensees to make facility changes without prior NRC approval

(3) initiate an optional performance-based, technology-inclusive regulation for nonlight-water reactors

(4) initiate a rulemaking to define high level performance-based instrumentation and control safety design principles and develop associated regulatory guidance that documents the acceptable standards that may be used to meet these principles

In addition to specific areas for transformative change, the team recommended specific actions and initiatives to foster and sustain transformation as a characteristic of the NRC's organizational culture. The Commission continues to consider the staff's recommendations in SECY-18-0060 in the context of the agency's broader transformation initiatives. Meanwhile, the staff continues to consider input from external stakeholders on other potentially transformational initiatives, such as revising the Reactor Oversight Process and RIDM efforts.

2.3.3 Major Regulatory Accomplishments

Since the issuance of the previous U.S. National Report in 2016, the NRC has achieved many regulatory accomplishments. The following are some of the major items:

- acceptance criteria for subsequent license renewal applications
- construction oversight
- decommissioning rulemaking
- implementation of Fukushima lessons learned
- issuance of new licenses
- small modular reactors and advance reactors licensing modernization

2.3.3.1 Acceptance Criteria for Subsequent License Renewal Applications

The standards for subsequent license renewal are identical to those for initial license renewal, as stated in 10 CFR 54.29, "Standards for Issuance of a Renewed License." In July 2017, the NRC issued NUREG-2192, which provides guidance to NRC staff reviewing a subsequent license renewal application to ensure that it meets the requirements of 10 CFR Part 54. NUREG-2192 ensures the quality and uniformity of NRC staff reviews and establishes a well-defined base from which to evaluate applicant programs and activities for the subsequent period of extended operation.

There was significant internal and external stakeholder involvement in the NRC's development of NUREG-2191 and NUREG-2192. Activities included the NRC staff review of the results from many aging management program audits; findings from an expert elicitation process that identified materials and components that could be susceptible to significant degradation during operation beyond 60 years; and the review of domestic and international operational experience, and public comments to identify technical issues that need to be considered for assuring the safe operation of NRC-licensed nuclear power plants.

Upon receipt of a subsequent license renewal application the NRC staff determines whether the applicant has provided the required administrative, technical, and environmental information. The staff uses the review checklist in NUREG-2192, Table 1.1-1, to determine whether the application is sufficiently complete and conforms to the requirements in 10 CFR Part 54.

To be considered reasonably complete and sufficient, the application should address each item in the checklist. When the NRC determines that the application is acceptable for review, the NRC staff begins its technical review. Additional information about the staff's subsequent license renewal activities can be found in Sections 2.3.1.9, 2.3.2.7, and Article 14.

2.3.3.2 Construction Oversight

The intent of the NRC's construction oversight program, as described by the staff in SECY-08-0155, "Update on the Development of the Construction Inspection Program for New Reactor Construction under 10 CFR Part 52," dated October 17, 2008, is to ensure that licensees and their contractors and vendors detect and correct problems in a manner that gives priority to quality and safety. The NRC has structured the construction inspection program and construction assessment process to reflect the rapidly changing nature of a construction environment. Additionally, the staff has evaluated NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants," dated May 1984; the lessons learned from domestic construction experience; and the lessons learned and challenges to the NRC's international regulatory counterparts currently overseeing nuclear construction activities. The NRC staff has incorporated these insights into construction inspection and assessment documents. As a result, the staff has developed a transparent and predictable process that objectively evaluates licensee performance of construction activities, and the effectiveness of licensee or contractor oversight and quality assurance efforts associated with construction. The NRC uses insights gained from assessing construction activities and insights from the annual construction oversight program self-assessment to improve its regulatory effectiveness. Construction oversight assessment reports are available on the NRC's public Web site at https://www.nrc.gov/reactors/new-reactors/oversight/crop.html.

The NRC has formed the Vogtle Readiness Group to identify and resolve any licensing, inspection, or regulatory challenges or gaps that could affect the schedule for completion of Vogtle Electric Generating Plant, Units 3 and 4. The Vogtle Readiness Group will provide high level assessments, coordination, oversight, and management direction of NRC activities associated with the licensing, inspection, testing, and operation of Vogtle Electric Generating Plant, Units 3 and 4. Section 2.3.1.7 of this report provides additional information on the staff's readiness to transition plants from construction to operations.

2.3.3.3 Decommissioning Rulemaking

The NRC is proposing to amend its regulations for the decommissioning of production and utilization facilities, although the issues considered by the NRC would apply mainly to nuclear power reactors.

As discussed in Section 2.3.1.8 of this report, the Commission directed the staff to proceed with a decommissioning rulemaking in SRM-SECY-14-0118. The NRC published an advance notice of proposed rulemaking in November 2015, and a regulatory basis document in November 2017. The staff used the regulatory basis to develop the proposed rule and on May 7, 2018, provided SECY-18-0055, "Proposed Rule: Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," to the Commission for its review and approval. The NRC staff has also developed draft implementation guidance documents for this rulemaking.

In several areas of the current regulations, there is no means to distinguish provisions that apply to a power reactor that has permanently ceased operations from provisions that apply to an operating power reactor. This rulemaking would amend the NRC's regulations in several areas to provide a regulatory framework for the transition to decommissioning. The NRC staff has recommended a graded approach that would align regulatory requirements with the reduction in radiological risk that occurs over time at a decommissioning facility. Additionally, because the current regulatory framework for decommissioning is adequate to protect public health and safety and the common defense and security, many of the new proposed requirements would be alternatives to current requirements.

The NRC staff has recommended regulatory changes to the Commission in the following 14 technical areas:

(1) emergency preparedness
(2) physical security
(3) cybersecurity
(4) drug and alcohol testing
(5) certified fuel handler definition and elimination of the shift technical advisor
(6) decommissioning funding assurance
(7) offsite and onsite financial protection requirements and indemnity agreements
(8) environmental considerations
(9) record retention requirements
(10) low-level waste transportation
(11) spent fuel management planning
(12) application of the backfit rule
(13) foreign ownership, control, or domination
(14) clarification of the scope of the license termination plan requirement

The Commission is currently evaluating the staff's recommendations in SECY-18-0055.

2.3.3.4 Implementation of Fukushima Lessons Learned

Since the March 2011 accident at Fukushima Dai-ichi, the NRC has made substantial progress in addressing the lessons learned from the accident and has implemented the most significant safety enhancements on or ahead of schedule. In 2011, the staff evaluated the lessons learned from the accident and prioritized its recommendations into three tiers based on the urgency of the action, the need for additional information, and the availability of critical skill sets. The most significant of these activities, referred to as Tier 1, were addressed by the issuance of orders, a request for information, and a rulemaking activity.

On March 12, 2012, the NRC issued three orders:

(1) EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events"

(2) EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation"

(3) EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents"

On March 12, 2012 the NRC also issued "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," to obtain information on current seismic and flooding hazard protection, seismic and flooding hazard reevaluations using up-to-date methods, and emergency preparedness communications and staffing capabilities.

On June 6, 2013, the NRC issued Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," which modified and superseded Order EA-12-050.

All U.S. operating power reactor licensees have completed the implementation of the safety enhancements required by the mitigation strategies and the SFP instrumentation orders. The staff has reviewed the licensees' required plans and strategies and completed onsite verification inspections. In SECY-16-0142, "Draft Final Rule: Mitigation of Beyond-Design-Basis Events," dated December 15, 2016, the staff proposed codifying the requirements of these two orders in the NRC's regulations. The Commission approved a final rule in SRM-SECY-16-0142, dated January 24, 2019. The final rule has been published and will become effective by September 2019.

All applicable operating power reactor licensees have implemented the safety enhancements required by the reliable hardened containment vent order. Verification inspections are ongoing and will continue through 2020. Because this order applies only to a limited group of plants (i.e., boiling-water reactors (BWRs) with Mark I or Mark II containments), the requirements did not need to be codified in NRC regulations.

All applicable U.S. operating power reactor licensees have completed the seismic and flooding related inspections and hazard reevaluations for the request for information. Licensees have implemented interim measures, if necessary, while additional evaluation of the impact of the reevaluated hazards on the sites is ongoing. The staff has reviewed the information provided and has identified those sites where additional evaluations of impact are needed. Some licensees needed to perform a flooding integrated assessment or a seismic PRA, while others needed to perform limited-scope evaluations. This determination was made based on the degree to which the new flooding or seismic hazard estimates varied from what was assumed during initial licensing. The NRC is using this additional information to determine if additional regulatory actions under the NRC's backfit process are warranted. Over 80 percent of applicable operating power reactor licensees have provided the requested information, with the remainder expected by December 2019.

Also in response to the request for information, all licensees completed assessments of their staffing and communication capabilities to effectively respond to multiunit and large scale emergencies. The NRC reviewed those assessments and performed inspections to verify the implementation and enhancements in conjunction with the postcompliance inspections for orders EA-12-049 and EA-12-051.

In 2015, the NRC staff reevaluated its plans to resolve the other staff recommendations (i.e., Tier 2 and 3 recommendations) in light of the safety benefit achieved through the Tier 1 recommendations and the direction given in related Commission decisions. The staff concluded, in general, that additional regulatory actions beyond those taken for the Tier 1 activities are not warranted.

The details of these activities and related documents can be found on the NRC's public Web site https://www.nrc.gov/reactors/operating/ops-experience/post-fukushima-safety-enhancements.html.

2.3.3.5 Issuance of New Licenses

The NRC has issued the following combined licenses since the submittal of the U.S. National Report in 2016:

- two combined licenses to Duke Energy Carolinas, LLC, on December 19, 2016, for William States Lee III Nuclear Station, Units 1 and 2
- two combined licenses for Duke Energy Florida, LLC, on October 26, 2016, for Levy Nuclear Plant, Units 1 and 2, which were later terminated upon Duke's request as shown in the list below
- one combined license to Virginia Electric and Power Company on June 2, 2017, for the North Anna Power Station, Unit 3
- two combined licenses to Florida Power & Light Company on April 12, 2018, for Turkey Point Nuclear Generating, Units 6 and 7

The following combined licenses were terminated since the submittal of the U.S. National Report in 2016:

- two combined licenses for Duke Energy Florida, LLC, on April 26, 2018, for Levy Nuclear Plant, Units 1 and 2
- two combined licenses for South Texas Project Nuclear Operating Company on July 12, 2018, for South Texas Project, Units 3 and 4
- two combined licenses for South Carolina Electric & Gas Company on March 6, 2019, for Virgil C. Summer Nuclear Station, Units 2 and 3

In total, in the United States there are eight combined licenses at five sites. The NRC has approved one design certification for Korea Hydro and Nuclear Power's APR1400 and has not approved any early site permits since the submittal of the U.S. National Report in 2016.

2.3.3.6 Advanced Reactors Licensing Modernization

As the NRC prepares to review and regulate a new generation of nonlight-water reactors, it has developed a vision and strategy to ensure the agency's readiness to effectively and efficiently conduct its mission for these technologies. The staff described the vision and strategy in its report, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," dated December 2016.

The NRC's vision and strategy for nonlight-water reactors has three strategic objectives: (1) enhancing technical readiness, (2) optimizing regulatory readiness, and (3) optimizing communication. The NRC prepared implementation action plans to identify the specific activities that it will conduct in the near-term (2016-2021), mid-term (2021-2026), and long-term (beyond

2026) timeframes to achieve nonlight-water reactor readiness. The NRC released the draft implementation action plans and sought stakeholder feedback. The NRC updated and finalized its implementation action plans to reflect stakeholder feedback in July 2017.

The staff issued SECY-19-0009, "Advanced Reactor Program Status," on January 17, 2019, which provides the status of the NRC staff's activities related to advanced reactors, including the progress and path forward on each of the implementation action plan strategies. It also summarizes the various external factors influencing the staff's preparations for possible licensing and deployment of advanced reactors.

As part of strategy 1 in the near-term implementation action plan, the NRC is acquiring and developing sufficient knowledge, technical skills, and capacity to perform regulatory activities. The NRC contracted with the Oak Ridge National Laboratory to conduct a 12-module training course on molten salt reactors in 2017. The NRC also contracted with Argonne National Laboratory to offer staff training on sodium-cooled fast reactors and high-temperature gas-cooled reactors in spring 2019.

As part of strategy 2 in the near-term implementation action plan, the NRC is acquiring and developing computer codes and tools to perform regulatory reviews. The NRC documented the status of efforts in these areas in a report, "Strategy 2 Near-Term Implementation Action Plan Progress Report for Fiscal Year 2017," dated December 7, 2017. The staff completed an initial screening of analysis codes for design-basis and beyond-design-basis event simulation, and identified a suite of tools for further examination and consideration. The staff had identified several functional areas to focus on in the near-term based on an assessment of the gaps between the current state and what is needed for advanced reactors. In addition, the staff conducted a phenomena identification and ranking table exercise for molten salt reactors.

As part of strategy 3 in the near-term implementation action plan, the NRC is optimizing its regulatory framework and licensing processes. In December 2017, the NRC issued "A Regulatory Review Roadmap for Non-Light Water Reactors," which describes potential examples of flexibility, including the use of a staged review process and conceptual design assessments during the pre-application period. The staff also issued RG 1.232, "Guidance for Developing Principal Design Criteria for Non Light Water Reactors," in April 2018. This activity is part of a joint initiative with DOE to address the general design criteria in Appendix A to 10 CFR Part 50, which the NRC developed primarily for light-water reactors, by adapting them to the needs of advanced reactor design and licensing. In January 2019, the Nuclear Energy Innovation and Modernization Act (NEIMA) was signed into law. This legislation includes provisions relating to a breadth of topics, ranging from budget formulation to advanced nuclear reactors. NEIMA requires the NRC to take a number of actions related to the licensing process for advanced reactors, including establishing via rulemaking a technology-inclusive regulatory framework for optional use by applicants for commercial advanced reactor licenses. The NRC has also engaged with the Licensing Modernization Project being led by Southern Company and coordinated by NEI, with costs shared with DOE. The Licensing Modernization Project's objective is to develop technology inclusive, risk-informed, and performance-based regulatory guidance for licensing nonlight-water reactors for the NRC's consideration and possible endorsement. On September 28, 2018, NEI submitted a proposed guidance document, NEI 18-04, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development." On May 3, 2019, the staff issued Draft Regulatory Guide (DG) 1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology To Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," for public comment. NEI-18-04 and DG 1353 provide a general methodology for identifying an appropriate scope and depth of information to be provided in applications to the NRC for licenses, certifications, and approvals for non-light water reactors.

As part of strategy 4 in the near-term implementation action plan, the NRC is facilitating the development of industry codes and standards needed to support the life cycle of nonlight-water reactors. The staff is participating in meetings for the development of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section III, Division 5, which provides rules for the design, construction, testing, certification, and quality assurance of high-temperature reactors and covers the use of metallic, graphite, and composite materials. The NRC has identified the 2017 Edition of this standard for potential endorsement to improve the efficiency and effectiveness of the agency's review process, provide designers with a stable set of rules for reactor development, and facilitate the certification of component vendors. The staff is also participating in several American Nuclear Society (ANS) standards meetings related to nonlight-water reactors.

As part of strategy 5 in the near-term implementation action plan, the NRC is identifying and resolving technology inclusive (not specific to a particular design or category) policy issues that affect regulatory reviews, siting, permitting, and/or licensing of nonlight-water reactors. The technology inclusive policy issues that the NRC staff has been discussing with stakeholders include the following:

- Siting. In November 2017, the NRC issued a draft white paper "Siting Considerations Related to Population for Small Modular and Non-Light Water Reactors," to facilitate stakeholder engagement with siting considerations associated with population distribution and density. SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors," dated February 7, 2016, identifies these considerations.

- Offsite Emergency Planning. On October 12, 2018, the NRC staff submitted SECY-18-0103, "Proposed Rule: Emergency Preparedness for Small Modular Reactors and Other New Technologies," to the Commission for review and approval. The proposed alternative emergency preparedness requirements adopt a consequence-oriented, risk-informed, and performance-based approach. In part, this rulemaking would reduce requests for exemptions from the current emergency preparedness requirements and promote regulatory stability, predictability, and clarity in the licensing process for these future facilities. Section 2.3.2.7 of this report discusses this rulemaking in more detail.

- Security and Safeguards Requirements. On December 14, 2016, NEI submitted a white paper, "Proposed Consequence-Based Physical Security Framework for Small Modular Reactors and Other New Technologies." This paper proposes an approach to security that considers the enhanced safety and security incorporated into these designs. NEI requested that the NRC establish regulatory positions on this approach and the associated policy and technical issues. The staff considered stakeholder input and issued SECY-18-0076, "Options and Recommendations for Physical Security For Advanced Reactors," on August 1, 2018. On November 19, 2018, the Commission approved the staff's recommendation to initiate a limited-scope revision of physical security regulations and guidance.

- Functional Containment Performance. On November 30, 2017, the NRC issued the draft white paper “Functional Containment Performance Criteria for Non-Light Water Reactor Designs,” to facilitate stakeholder engagement on the use of a functional containment approach. The staff considered stakeholder feedback and issued SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water Reactors,” on September 28, 2018. On December 4, 2018, the Commission approved the staff’s proposed methodology.

As part of strategy 6 of the of near-term implementation action plan, the NRC is optimizing external communications. The NRC is conducting public meetings with stakeholders every 4 to 6 weeks. The agency continues to meet with potential applicants, upon request, and to share information with various international groups and counterparts, including NEA, the International Atomic Energy Agency (IAEA), and the Generation IV International Forum. The NRC also chairs NEA’s Working Group on the Safety of Advanced Reactors for international regulators of nonlight-water reactors, which brings interested regulators together to discuss common interests, practices, and problems and to address their regulatory interests and research needs.

2.4 International Peer Reviews and Missions

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

2.4.1 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 2016 U.S. National Report at the seventh CNS review meeting in March 2017 were very positive.

2.4.1.1 Items Resulting from the Contracting Parties’ Peer Review

A review of the questions raised by other contracting parties on the 2016 U.S. National Report identified the following areas of interest:

- emergency preparedness
- Fukushima lessons learned
- good practices and areas of good performance
- license renewal and aging
- licensing
- quality assurance
- Reactor Oversight Process
- risk-informed regulations and PRA
- safety culture
- Vienna Declaration on Nuclear Safety

The NRC’s presentation during the 2017 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 had not implemented any good practices in the last review cycle. Good practices are defined as follows:

> a new or revised practice, policy or programme that makes a significant contribution to nuclear safety. A Good Practice is one that has been tried and proven by at least one Contracting Party but has not been widely implemented by other Contracting Parties; and is applicable to other Contracting Parties with similar programmes

Areas of good performance are defined as the following:

> a practice, policy or programme that is worthwhile to commend and has been undertaken and implemented effectively. An Area of Good Performance is a significant accomplishment for the particular CP [contracting party] although it may have been implemented by other CPs

The group participants identified the following areas of good performance by the United States:

- making extensive use of systematic and comprehensive operating experience programs and processes
- implementing Project Aim, which focuses on the safety mission and is aimed at prioritizing activities and improving the NRC's efficiency, effectiveness and adaptability
- offering extensive opportunities for public engagement in the regulatory processes
- summarizing the changes in the U.S. National Report in a table format and including a revision bar to facilitate the peer review process
- using a systematic approach to prepare staff for all phases of the nuclear power reactor life cycle
- using risk considerations in regulatory oversight in categorization and treatment of SSCs
- conducting safety culture self-assessment at the sites every 2 years
- issuing RG 5.74, "Managing the Safety/Security Interface," Revision 1, which includes cybersecurity as part of the safety and security assessment, in April 2015

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

- establishing the acceptance criteria for life extension beyond 60 years (discussed in Section 2.3.3.1 of this report)
- clarifying backfitting guidance and implementation (discussed in Section 2.3.2.3 of this report)
- changes in the demographics, experience, and knowledge of regulatory body staff (discussed in Section 8.1.6.2 of this report)

- ensuring continuity during the oversight transition from construction to operation (discussed in Section 2.3.1.7 of this report)

The current U.S. National Report addresses these issues in the sections mentioned above.

2.4.1.2 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. CNS contracting parties have led some of the most important efforts, 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 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, representatives decided to continue moving the Convention forward by recommitting and rededicating the Nations 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 IAEA Information Circular (INFCIRC) 872, dated February 18, 2015, states the following:

- 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 offsite 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 United States has consistently addressed the principles documented in the Vienna Declaration on Nuclear Safety since the inception of the CNS. To facilitate the contracting parties' peer review, the NRC has included in this report a 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 CNS articles.

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, 21, 30, 40, 50, 52, 55, 70, 73, and 100. These NRC regulations govern the design, siting, construction, and operation of nuclear power plants and serve to prevent accidents and mitigate adverse consequences in a way 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 by preventing 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. Accidents are prevented and mitigated through the establishment of criteria for control and safety systems, such as the containment, reactor coolant systems, and emergency core cooling systems. The regulatory objectives and measures include the following:

- Robustness of Defense-in-Depth. The defense-in-depth philosophy is a fundamental element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The philosophy ensures that safety will not wholly depend on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.

  Defense-in-depth embraces a broad set of principles and requirements, including: (1) the need to prevent accidents from occurring and to mitigate 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 are addressed by requirements such as the "single failure" assumption, and (4) siting new nuclear power plants in lower population areas and areas with natural characteristics that are less adverse than other possible locations. Section 18.1 of this report provides additional details about the NRC's defense-in-depth philosophy.

- Prevention of Accidents. Prevention of accidents is normally considered the first layer of defense-in-depth. Accidents are prevented by conservative design and high quality and standards in construction and operation. The NRC governs these aspects through its regulations and programs, including, but not limited to, general design criteria for the design of SSCs in Appendix A to 10 CFR Part 50; quality assurance requirements in Appendix B, "Quality Assurance Criterial for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; industry codes and standards required by regulation or endorsed for use by the NRC; and the NRC's programs for inspecting design, construction and operational activities and enforcing compliance with its regulations. The general design criteria govern the design of multiple fission product barriers, protection and reactivity control systems, fluid systems, containment design, and fuel and radioactivity control.

- Beyond-Design-Basis Events. Since the accident at Three Mile Island in 1979, the NRC has implemented requirements for prevention and mitigation of accidents not included in the original design bases for light-water reactors. On August 8, 1985, the Commission published its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," (50 FR 32138). This statement describes the policy that the Commission intended to use to resolve safety issues related to a reactor accident more severe than design basis accidents.

Several important examples of regulations that address beyond-design-basis events include anticipated transients without scram, station blackout, loss of large areas of the plant because of fires and explosions, and mitigation strategies for beyond-design-basis external events. New plants are also required to (1) meet analysis and design requirements aimed at protecting key barriers against release or radioactivity (i.e., fuel, reactor vessel, and containment) from the impact of a large commercial aircraft on the plant and (2) perform a PRA for their proposed design. The PRA is not limited to modeling and analyzing design basis accidents; the PRA models and analyzes all potential severe accidents contributing to core damage and radionuclide releases.

The NRC 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. However, the United States has not relied on, nor will it rely on in future nuclear power plant licensing, an unusually remote location to ameliorate what would otherwise be considered unacceptable radiological risks of either early radioactive releases or long-term offsite contamination from a proposed plant. 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 protecting 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.

The NRC requires reactor licensees to establish emergency plans that implement the U.S. Environmental Protection Agency (EPA) protective action guidelines to mitigate radiological effects in the unlikely event of a reactor accident capable of a large release of radioactive material. The NRC also requires 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 event of a radioactive release. EPA has established dose based protective action guidelines (https://www.epa.gov/radiation/protective-action-guides-pags) for the relocation and reentry of members of the public during the intermediate phases of a radiological incident or accident. In addition, in February 2009, DOE published "Preliminary Report on Operational Guidelines Developed for Use in Emergency Preparedness and Response to a Radiological Dispersal Device Incident" (see DOE/HS-0001; ANL/EVS/TM/09-1, at http://www.evs.anl.gov/resrad/documents/ogt_manual_doe_hs_0001_2_24_2009c.pdf). These guidelines, which provide stay times and concentrations for several different sets of assumptions about the exposure, can be used to calculate doses to members of the public.

The Second Principle of the 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 regularly scheduled baseline and targeted inspections, 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.

One of the many inspections that the NRC conducts is in the area of problem identification and resolution. This inspection, which is largely governed by 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," focuses on correcting conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances for those SSCs subject to 10 CFR Part 50, Appendix B. As needed safety improvements are identified and imposed, deadlines for licensee implementation are established. Conditions need to be corrected in a manner commensurate with their safety or security significance, but the time for correction should not exceed one operating cycle unless justified to the NRC by licensee senior management. An example would be the post-Fukushima requirements for certain designs to make various safety improvements within time periods specified in the order. Section 2.3.3.4 of this report discusses the NRC post-Fukushima orders and accomplishments.

As part of the safety review, the staff uses the Office of Nuclear Reactor Regulation Office Instruction LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," Revision 4, dated June 2, 2014, to outline the process by which the staff and managers evaluate and communicate risk-informed decisions and thereby improve the NRC's efficiency and effectiveness. Also, "backfitting" is the process by which the NRC determines whether to issue new or revised requirements or staff positions interpreting those requirements to licensees of nuclear power reactor facilities. Backfitting is done only after formal, systematic review to ensure that changes are properly justified and suitably defined. Two NRC rules, 10 CFR 50.109, "Backfitting," and 10 CFR 50.54(f) provide the requirements for proper justification of backfits and information requests, respectively. Sections 2.3.2.3 and 14.1.5.2 of this report present additional information about the backfit process.

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 Sections 6.3.5 and 6.3.11 of this report. The NRC communicates information internally to ensure that the technical staff can factor operating experience into its reviews of plant safety. The NRC staff communicates with INPO to ensure that relevant operating experience reviewed by the industry is also considered in NRC reviews. The NRC communicates through the issuance of generic communications to share its operating experience insights with the industry, the public, and the international community. In addition, the staff can revise inspection procedures when operating experience indicates 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.

To a large extent, the international community conducts comprehensive periodic safety reviews at set intervals 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 safety inspections, audits, license renewals, and assessment programs that deal with specific safety and aging issues, significant events, and changes in safety standards and practices as they arise, to provide comprehensive review and oversight. 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 agency's regulatory approach of promptly addressing new information when it is discovered and promptly taking appropriate regulatory action, rather than awaiting a periodic review.

The IAEA IRRS mission and followup mission conducted in 2010 and 2014, respectively, evaluated the effectiveness of the NRC's regulatory approach. During the 2010 IRRS mission, the NRC correlated its regulatory programs to the 14 periodic safety review "safety factors" to demonstrate that the NRC programs robustly meet the intent of the periodic safety review. The IRRS team concluded that the NRC has 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 meeting regulatory requirements. Sections 8.1.5.2 and 14.1.5 of this report further discuss the results of the IRRS mission and the alternative program that the United States uses in lieu of conducting periodic safety reviews.

<u>The Third Principle of the Vienna Declaration on Nuclear Safety</u>. The NRC's regulatory requirements and guidance documents undergo systematic reviews and revisions, which are informed by international standards and guidance documents. Built into the process for updating the NRC's guidance is an examination of applicable technical basis information, including related guidance available in domestic and international consensus standards, IAEA nuclear safety standards and recommendations, and other relevant documents. NRC RGs, for example, routinely cite or reference relevant IAEA safety standards and guides that address similar technical content and note that the IAEA safety standards present international good practices to help users striving to achieve high levels of safety. The NRC's RGs state that they are consistent with the basic safety principles in the cited IAEA documents.

Also, NRC senior managers serve as the U.S. delegates to each of the five safety standard committees under the aegis of the IAEA Commission on Safety Standards. This participation helps harmonize NRC requirements and guidance with international standards and guidance. Section 8.1.5.1 of this report provides additional information about how the NRC uses IAEA safety standards.

2.4.1.3 Areas of Focus for the Eighth Convention on Nuclear Safety

During the 2017 CNS review meeting the contracting parties agreed to continue to hold topical sessions during the review meetings. Contracting parties were invited to propose recommendations for the topical sessions to be held at the eighth CNS review meeting. In October 2018, the contracting parties agreed on the following areas of focus for these sessions.

<u>Aging Management</u>. The NRC continues to give special focus to issues associated with aging management and license renewal. The NRC has also received requests to review subsequent license renewal applications, which would allow licensees to operate plants up to 80 years. The Commission concluded that the current regulatory framework for the first license renewal is sound and sufficient to provide reasonable assurance that the power reactors can safely operate beyond 60 years. SRM-SECY-14-0016 identified four technical issues related to subsequent license renewal including the following: reactor pressure vessel neutron embrittlement at high fluence, irradiation assisted stress corrosion cracking of reactor internals

and primary system components, concrete and containment degradation, and electrical cable qualification and condition assessment. Section 2.3 and Article 14 of this report provide additional information on aging management, license renewal, and subsequent license renewal.

Safety Culture. 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 agency also leads by example by fostering a culture in which all employees may live the NRC's values and adhere to the Principles of Good Regulation to support the mission to protect public health, safety, and the environment. Section 10.4 of this report presents more information on safety culture.

2.4.2 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.

2.4.3 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 2017, Sequoyah Nuclear Plant hosted an OSART mission. The OSART team concluded that the managers and the staff of Sequoyah are committed to improving the operational safety and reliability of their station. A followup OSART mission was hosted in April 2019, as discussed in greater detail in Section 8.1.5.3 of this report. The next OSART in the United States is tentatively scheduled to take place in 2020.

PART 2
Article-by-Article Reporting

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, and generic communications. 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 radioactive material 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.

The agency achieves its strategic safety goal by ensuring that licensee performance is at 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 five performance goals and indicators in this Annual Performance Plan, which are discussed in NUREG-1100, Revision 1, "Congressional Budget Justification: Fiscal Year 2020," Volume 35, dated March 2019. These goals and indicators are used to track the effectiveness of the NRC's nuclear safety regulatory programs and determine whether the strategic safety goal has been met. Of these five, the following four indicators are related to commercial nuclear power plants:

(1) number of radiation exposures that meet or exceed abnormal occurrence[7] criterion 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

7 All references to the abnormal occurrence criteria in this section refer to the criteria approved by the Commission in SRM-SECY-17-0019, "Final Revision to Policy Statement on Abnormal Occurrence Reporting Criteria."

(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.E

In FY 2018, the NRC met all its performance indicator targets, and thus, achieved its strategic safety goal objective. The NRC also met its previous performance indicators in FYs 2016 and 2017.

6.2 Nuclear Installations in the United States

Appendix B to this report lists all operating nuclear installations in the United States, as discussed in NUREG-1350, Volume 30, “Information Digest 2018-2019,” dated August 2018. Since the issuance of NUREG-1350, Volume 30, Oyster Creek Nuclear Generating Station and Pilgrim Nuclear Power Station have ceased operations bringing the total to 97 operating nuclear installations in the United States. Eleven additional nuclear installations have announced an intent to permanently cease operations in the next few years. Section 2.3.1.8 of this report provides additional information on the staff’s readiness to transition plants from operation to decommissioning status.

Appendix A to NUREG-1350 also lists installations in the United States that are under active construction or deferred plant status. Bellefonte Nuclear Station, Units 1 and 2, is currently in deferred status per the “Commission’s Policy Statement on Deferred Plants,” dated October 14, 1987 (52 FR 38077). The NRC issued the combined licenses for the Vogtle Electric Generating Plant, Units 3 and 4, in February 2012. These two AP1000 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. Additional information on the NRC’s construction oversight activities and the staff’s readiness to transition plants from construction to operation status can be found in Sections 2.3.3.2 and 2.3.1.7 of this report, respectively.

6.3 Regulatory Processes and Programs

6.3.1 Reactor Licensing

To construct and operate a new nuclear reactor, an entity must apply to the NRC for a license. After accepting the application, the NRC staff will conduct a safety and environmental review. The public has opportunities to participate through a hearing process. The NRC licensed all currently operating nuclear plants under the two-step process, specified in 10 CFR Part 50, first issuing a construction permit and then an operating license. Since 1976, the NRC has not received any applications to construct a new reactor under 10 CFR Part 50.

The revised, single-step process adopted in 1989, is specified in 10 CFR Part 52 and provides direction for issuing a combined license for construction and operation of a new reactor. The NRC has issued 14 combined licenses since 2012, authorizing the construction and operation of 14 new units at eight nuclear power plant sites in the United States. Six of the licenses at three sites were subsequently terminated at the licensees’ request. Eight licenses at five sites remain in place. Since the issuance of the last U.S. National Report, the NRC has approved the combined licenses for William States Lee III Nuclear Station, Units 1 and 2; North Anna, Unit 3;

and Turkey Point, Units 6 and 7. The NRC also issued a license for the Levy Nuclear Plant, Units 1 and 2; however, this license was subsequently terminated as stated above. Currently there are no combined license applications under review.

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 six design certifications and three design certification amendments. In April 2019, the NRC issued a direct final rule certifying the Korea Hydro and Nuclear Power's APR1400. Two applications are currently under review: (1) Mitsubishi's U.S. Advanced Pressurized-Water Reactor (U.S. APWR) and (2) NuScale Power's small modular reactor. In addition, NRC is also actively reviewing the General Electric Hitachi Advanced Boiling-Water Reactor (ABWR) renewal application.

As specified in 10 CFR Part 52, the NRC can 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. The staff has not approved any new early site permits since the issuance of the last U.S. National Report. One early site permit application for the Clinch River Nuclear Site is currently under review. Article 18 and 19 of this report provide 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 or combined 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). Articles 14, 17, and 18 of this report contain additional information.

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 regulatory authority to take actions necessary to protect public health and safety and the environment and may order immediate licensee actions, up to and including a plant shutdown, to address unacceptable safety or security performance at a domestic nuclear power plant.

The Reactor Oversight Process monitors licensee performance in three strategic performance 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 site to monitor plant status, perform routine inspections, and respond immediately 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 and performance-based baseline inspections, which represent the level of NRC inspection required to adequately assess licensee performance. Baseline inspections are used in conjunction with performance indicator data, which are 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 assesses 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 evaluate 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 of nuclear power plant licensees and to ensure that trends in nuclear industry and licensee performance are appropriately addressed. After each Agency Action Review Meeting, the NRC informs licensees of any 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 annual public meetings. Performance information and additional information about the Reactor Oversight Process can be accessed at https://www.nrc.gov/reactors/operating/oversight.html.

As of August 1, 2019, the Action Matrix assessment of licensee performance at nuclear reactors was as follows:

- Column 1: 92 reactor units in Licensee Response
- Column 2: 5 reactor units in Regulatory Response
- Column 3: No units in Degraded Cornerstone
- Column 4: No units in Multiple/Repetitive Degraded Cornerstone
- Column 5: No units in Unacceptable Performance

The results of annual Reactor Oversight Process self-assessments indicate that the program remains effective. SECY-19-0037, "Reactor Oversight Process Self-Assessment for Calendar Year 2018," dated April 12, 2019, documents the most recent status of the NRC's self-assessment program. The staff continues to evaluate and implement program improvements based on lessons learned and feedback from stakeholders and independent assessments, consistent with the continuous improvement focus of the Reactor Oversight Process.

The Reactor Oversight Process has developed into a mature oversight program since its inception in 2000 and has been a model followed by several countries. The NRC 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. The NRC staff is currently working on an initiative to further enhance the process. This initiative originated, in part, from the NRC's transformation initiative. Through the transformation initiative, the staff received 72 recommendations on the Reactor Oversight Process were submitted by the NRC staff, and an additional 27 recommendations were submitted by NEI on behalf of the nuclear industry. Feedback from internal and external stakeholders indicate that the oversight framework and the Reactor Oversight Process goals and objectives remain sound and effective; however, stakeholders did identify potential improvements. The staff will evaluate and address these recommendations. Section 2.3.2 of this report provides additional information on changes to the Reactor Oversight Process and transformation at the NRC.

6.3.3 Industry Trends Program

The NRC discontinued the Industry Trends Program in 2016 because of Project Aim, which was the NRC's effort to develop an integrated prioritization and re-baselining of agency activities. The agency determined that, while the Industry Trends Program provided data that helped validate broad industry performance trends, no regulatory action resulted from its insights. Any negative trends in industry performance that the Industry Trends Program could have highlighted 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 NRC created the Accident Sequence Precursor Program in response to the insights and recommendations of NUREG-75/014 (WASH-1400), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975, and the 1979 accident at Three Mile Island Nuclear Station, Unit 2. This 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 the core damage risk; provides a partial check on dominant core damage scenarios predicted by PRAs; and provides feedback to regulatory activities.

The Accident Sequence Precursor Program supports the NRC's safety and performance objectives, strategies and goals. Its objectives include: (1) evaluating operating events, and trends, and advances in science and technology for safety implications to enhance the regulatory framework, (2) assist in preventing, mitigating and responding to accidents, (3) assist in preventing accident precursors and reductions of safety margins that are of high risk significance, (4) providing feedback to improve the NRC Standardized Plant Analysis Risk models, (5) increasing NRC and licensee staff knowledge to improve PRA models by discussing and reviewing key modeling issues, including implementation of PRA standards with licensees, and (6) communicating risk significant insights to licensees for incorporation into their operating experience, corrective actions, or plant improvement programs.

To identify potential precursors, the NRC reviews plant events from licensee event reports (LERs) and inspection reports. The staff then analyzes any identifies potential precursors by calculating the probability of an event leading to a core damage state. A plant event can be one of two types: (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, characterized 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 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 latest program results, trend analyses, and insights are documented in the "U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program 2018 Annual Report," dated April 2019. The report provides the following insights for the 2009–2018 period:

- The program identified 138 precursors, with 62 percent leveraging the results of the significance determination process.

- The program identified no significant precursors. The last significant precursor was identified in 2002, involving concurrent degraded conditions at Davis Besse Nuclear Power Station.

- A statistically significant decreasing trend was identified for all precursors. In addition, the program found decreasing trends for precursors resulting from initiating events, degraded conditions, losses of offsite power, higher-risk precursors, and precursors occurring at PWRs. No statistically significant trends were identified for precursors occurring at BWRs and precursors resulting from failures of emergency diesel generators.

- The program identified seven precursors with a conditional core damage probability or an increase in core damage probability greater than or equal to 1×10^{-4} in 2010–2012; however, none have been identified since.

- Precursors involving degraded conditions (90 precursors) outnumbered initiating events (48 precursors). Approximately one third of initiating event precursors resulted from natural phenomena (e.g., severe weather, seismic event). Of the 90 degraded condition precursors, 26 percent existed for at least 10 years.

A review of the Accident Sequence Precursor Program data and trends for the past decade indicates the following:

- Current agency oversight programs and licensing activities remain effective as shown by decreasing trends in the occurrence rate of all precursors and overall risk from precursors as shown by the integrated Accident Sequence Precursor index.

- Licensee risk management initiatives are effective in maintaining a flat or decreasing risk profile for the industry.

- There are no indications of increasing risk from the potential cumulative impact of risk-informed initiatives.

- No new component failure modes or mechanisms have been identified, and the likelihood and impacts of accident sequences have not changed.

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.

Most recently, the NRC has established a center of expertise integrating the functions of the Construction Experience Program (described in Section 18.4 of this report) and the Operating Experience Program (described in Section 19.7 of this report) as a measure to address operating experience more efficiently.

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, in other reports submitted under licensee reporting requirements, and 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. The NRC staff systematically screens operating experience for safety significance and generic implications. The staff also determines the need for further action and application of lessons learned from plant operating experience.

Operating experience also plays a key role in the development and application of NRC nuclear plant risk models, which themselves, are an integral component in the agency's risk-informed regulatory environment. The NRC obtains additional operational data via a longstanding industry-led program managed by INPO, which provides key component and system operational, test and failure data to the NRC. The information is analyzed, made available to the public, and incorporated into an NRC risk model for each nuclear plant, which may then be used to evaluate potential areas of concern identified in licensee performance.

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 its 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, including document collections, contacts, search tools, sources, and reference material. The agency's public Web site at http://www.nrc.gov/reading-rm/doc-collections/event-status/ contains all of the event reports that licensees have submitted to the NRC.

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 for safety or security that cannot be more appropriately addressed by other regulatory programs or processes. Proposed generic issues originate from safety evaluations, operational events, and suggestions from NRC staff members, outside organizations, or members of the public. For emergent issues, the NRC uses LIC-504 to evaluate whether immediate actions are needed. Actions may include issuing orders requiring plants to make changes or shut down, if necessary.

The Generic Issues Program consists of three stages: screening, assessment, and regulatory office implementation. A review panel, consisting of NRC staff with appropriate skill sets, determines 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 the seven screening criteria:

(1) significantly affects public health and safety, security, or the environment

(2) applies to two or more facilities

(3) is not currently being addressed through other NRC 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 in a timely manner

(6) is well defined and discrete

(7) may involve review, analysis, or action by the licensee

If the review panel finds that the proposed issue meets all the screening criteria, it proceeds 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., INs, bulletins, or GLs) and, if necessary, issuing orders, and initiating a rulemaking.

The Generic Issues Program staff tracks the status of the generic issue until all required actions are taken and the issue is closed. Additional information on the Generic Issues Program appears on the NRC public Web site at http://www.nrc.gov/about-nrc/regulatory/gen-issues.html; a history of generic issues appears in NUREG-0933, "Resolution of Generic Safety Issues."

6.3.7 Rulemaking

The NRC's rulemaking process is used to issue new or revised requirements that licensees must meet to obtain or retain a license or certificate to use nuclear materials or to operate a nuclear facility. The NRC may pursue a rulemaking based on a congressional mandate, an Executive Order, a petition for rulemaking from outside the NRC, or an internal recommendation from the NRC staff. To ensure early Commission engagement before expending significant NRC staff resources on any rulemaking, the NRC staff is required 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) on the new rulemaking activity. The staff may ask the Commission that a rulemaking activity be discontinued or delayed at any stage in the rulemaking process.

The NRC invites a diverse body of stakeholders to participate in the agency's rulemaking process. These stakeholders include the public, Congress, other Federal agencies, States, local governmental bodies, Indian Tribes, industry, technical societies, and citizen groups. The NRC seeks public involvement during the rulemaking process to understand and address any stakeholder concerns. The agency may publish related documents, such as an advance notice of proposed rulemaking and a draft regulatory basis, early in the rulemaking process to seek public comment.

In addition, any member of the public may petition the NRC to develop, change, or rescind a rule under 10 CFR 2.802, "Petition for Rulemaking—Requirements for Filing." If the petition for rulemaking meets the NRC's requirements for docketing, then the NRC publishes a notice of docketing of the petition in the *Federal Register*. 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 or deny the petition (in its entirety or in part). Section 8.1.7 of this report provides more information on the tools that the NRC uses to ensure openness and transparency in its work.

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 to the rule, and forwards the final rule for Commission approval and publication in the *Federal Register*.

The NRC manages its rulemaking dockets using the Federal Docket Management System, a tool used across the Federal Government that provides a single point of access at http://www.regulations.gov. Through this Web site, the public can access thousands of documents related to NRC rulemaking actions from May 1996 to the present. The Web site contains proposed and final rules that have been published in the *Federal Register* along with any comments received, petitions for rulemaking, and other types of documents related to the rulemaking process.

All documents referenced within each rulemaking are also made available to the public for inspection and comment during the public comment periods. These documents are made available in several ways to ensure that the public has the information needed to understand and participate in the rulemaking. For referenced agency records, the public can easily search the NRC's official records by using the NRC's Agencywide Documents Access and Management System (ADAMS). The NRC also ensures that all documents related to rulemakings are available in the NRC's Public Document Room.

Rulemaking authority for the NRC is vested in the Commission. The NRC's Executive Director for Operations is authorized to approve final rules that have been delegated by the Commission, including those 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., those that have a significant impact on the economy) become effective at least 60 days after the date of publication.

6.3.8 Fire Protection Regulation Program

The NRC has two main focuses in fire protection regulation: (1) implementation of the risk-informed, performance-based fire protection rule (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,” in December 2009, and NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements,” in September 2010. These documents reflect lessons learned from the pilot application reviews. In June 2010, the NRC approved the first risk-informed fire protection program for Shearon Harris Nuclear Power Plant. As of July 2019, the agency has approved a total of 28 risk-informed fire protection programs that represents 44 reactors. Nuclear power plants that are not transitioning or have not completed their transitions to the risk-informed, performance-based fire protection rule are regulated under their current, deterministic licensing bases.

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. Plants that are not transitioning, or have not completed their transition, to 10 CFR 50.48(c) are inspected under Inspection Procedure (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),” also dated January 31, 2013. Findings identified for licensees under both regulatory frameworks are evaluated using Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix F, “Fire Protection Significance Determination Process,” dated May 2, 2018. Fire protection enforcement discretion has ended for sites not transitioning to 10 CFR 50.48(c).

RG 1.189, “Fire Protection for Nuclear Power Plants,” Revision 2, dated 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 worked with industry stakeholders to enhance guidance on fire-induced multiple spurious operations through the development of Volume 3 to NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE),” dated November 2017. JACQUE-FIRE, Volume 3, builds upon the two prior volumes of that report to provide a better understanding of failure modes that might occur in electrical control circuits of nuclear power plants because of fire damage to electric cables. This report documents progress in resolving longstanding issues related to evaluation of multiple spurious operations and deterministic postfire safe-shutdown analysis. Specifically, the report provides a more consistent application in multiple topical areas:

- clarification of circuit failure modes and terminology
- recommendations for revising phenomena identification and ranking table panel positions and findings
- technical design considerations for shorting switch applications

- recommendations for evaluation of combinations of hot short-induced multiple spurious operations

- recommendations for the duration of hot short-induced spurious operations in direct current (dc) and alternating current (AC) power control circuits for deterministic postfire safe-shutdown analysis

- disposition of secondary fires due to a fire-induced open circuited current transformer

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 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, 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 as the nuclear facility transitions from "operation" to "decommissioning" status. When the end of the decommissioning process nears, the licensee can apply to terminate its license and release the site from regulatory control. 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 in accordance with Subpart E 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 1997, the NRC added requirements in 10 CFR 20.1406, "Minimization of Contamination," for descriptions of 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. This regulation integrated with the concept of dose optimization to a level as low as reasonably achievable (ALARA) in the design criteria for new facility construction. The design criteria require 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. New applicants use the guidance in RG 4.21, "Minimization of Contamination and Radioactive

Waste Generation: Life-Cycle Planning" dated June 2008, to strengthen decommissioning by minimizing contamination and dose optimization (i.e., ALARA).

In 2011, the NRC issued the Decommissioning Planning Rule, which updated 10 CFR 20.1501, "General." RG 4.22, "Decommissioning Planning during Operations," dated December 2012, contains guidance for implementing the rule. To strengthen future decommissioning financial assurance requirements and prevent future legacy sites at existing operating and decommissioning facilities, 10 CFR 20.1501 requires all licensees to perform surveys, including of the subsurface, near sources of potential leaks to provide early detection of the release of radioactive materials to the environment. Identification of a leak would require the licensee to either repair the leak and remediate the area promptly or provide additional decommissioning funding to remediate contamination before license termination.

The regulations pertaining to decommissioning funds for commercial power reactors are in 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning" and 10 CFR 50.82. The licensees must provide reasonable assurance that funds will be available for the decommissioning process. A power reactor licensee operating under a 10 CFR Part 50 or 10 CFR Part 52 license may use a prepaid segregated fund, external sinking fund, surety, insurance or guarantee, a statement of intent, contractual obligation, or a combination of these methods, which are described in 10 CFR 50.75(e)(1)(i-v). A power reactor licensee may propose other methods of assurance but, to obtain NRC approval, must show that the method is equivalent to the methods listed in the NRC's regulations. Electric utility licensees use external sinking funds to collect their decommissioning funds, while nonelectric utility plant licensees default to using a discounted prepayment method for decommissioning funding. NUREG-1577, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1, dated December 2001, and RG 1.159, Revision 2, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," dated, October 2011, present additional guidance on power reactor licensee methods of providing 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. 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 to allow unrestricted use of the sites. However, licensees transitioning from operations to decommissioning require several license amendments and exemptions from current NRC regulations to reduce requirements commensurate with their decommissioning status. As discussed in Sections 2.3.1.8 and 2.3.3.3 of this report, the Commission directed the NRC staff to conduct a rulemaking to provide a safe, effective, and efficient decommissioning process by, among other things, reducing the need for requests for license amendments and exemptions from existing regulations; addressing the appropriateness of maintaining the existing options for decommissioning and the 60-year timeframe for decommissioning; evaluating the need for a licensee's Post-Shutdown Decommissioning Activities Report to be approved by the NRC; and assessing the appropriate role of States, local governments, and nongovernmental 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 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 role like a technical support organization in other countries. In conducting the Reactor Safety Research Program, the NRC anticipates the challenges posed by the introduction of new technologies. The NRC also continues to seek 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, where such activities do not compromise NRC's independent regulatory decisionmaking. The agency also continues to offer 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 October 14, 2008, Commission's "Policy Statement on the Regulation of Advanced Reactors," (73 FR 60612) 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 expects will arise during its review of advanced reactor designs.

6.3.11 Generic Communications and Orders

Generic communications are the NRC's primary method of communicating a common need for information or an approach to resolve an issue, or communicating the NRC's position and information on issues pertaining to a matter of regulatory interest. Generic communications also allow the NRC to communicate and share industry experiences and send information to specific classes of licensees and interested stakeholders.

The following are several types of generic communications:

- Bulletins. Bulletins typically contain urgent requests for information or actions in the NRC's regulatory arena and typically require responses.

- Generic Letters. GLs typically request information or actions in the NRC's regulatory arena and typically require responses.

- Regulatory Issue Summaries. Regulatory issue summaries (RISs) typically communicate or clarify NRC technical or policy positions on regulatory matters or request voluntary participation, which will assist the NRC in the performance of its functions.

- Information Notices. INs transmit information focused on operational events or analytical experience.

- Information Assessment Team Advisories. These advisories provide urgent, time-sensitive, threat-related information to specified licensees.

- Security Advisories. These advisories communicate emergent, timely, operational or situational awareness threat-related information about the security and common defense of national infrastructure under the NRC's cognizance. They are operational in nature and issued in response to an urgent situation or recently identified vulnerability.

The NRC has extensive experience using the generic communications program. A relevant example is its response to the Fukushima accident, when the staff used generic communications to inform and request information from licensees to confirm that there were no imminent safety concerns at U.S. nuclear facilities.

On March 18, 2011, the NRC issued IN 2011-05, "Tōhoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants." This IN informed U.S. operating power reactor licensees and applicants of the effects from the earthquake and tsunami.

On May 11, 2011, the NRC issued Bulletin 2011-01, "Mitigating Strategies," which required licensees to provide a comprehensive verification of their compliance with the regulatory requirements of 10 CFR 50.54(hh)(2), as well as provide information associated with the licensee's mitigation strategies under that section.

In 2012, the NRC asked U.S. nuclear power plant licensees to perform detailed inspections of their currently installed seismic and flooding protection features to ensure that the features met current requirements, and to identify, correct, and report any degraded conditions. During the flooding walkdowns, a number of sites found deficiencies and took corrective actions to remedy them. On January 9, 2015, the NRC issued IN 2015-01, "Degraded Ability To Mitigate Flooding Events," to inform licensees of the external flood protection deficiencies.

On June 22, 2015, the NRC issued GL 2015-01, "Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities," to address concerns with fuel facilities' safety assessments and the supporting documentation with respect to the treatment of natural phenomena hazards. The letter also sought information needed to help determine the applicability of the lessons learned to facilities other than operating power reactors.

Another important regulatory tool is the NRC's Enforcement Program, which allows the agency to issue orders to modify, suspend, or revoke a license; to cease and desist from a given practice or activity; or take other necessary action. For example, as part of the response to the Fukushima accident, the NRC quickly determined that no imminent safety issue existed and no nuclear power plants were required to shut down. However, on March 12, 2012, the NRC issued three immediately effective orders to operating power reactor licensees and construction permit holders requiring them to take critical actions.

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. Section 2.3.3.4 of this report provides additional information on the implementation of the Fukushima-related orders, responses to information requests, and inspections. Section 9.3 of this report discusses the Enforcement Program and tools the NRC uses to ensure that licensees meet their primary responsibility to maintain safety.

6.4 Vienna Declaration on Nuclear Safety

On February 18, 2015, the contracting parties to the CNS issued the Vienna Declaration on Nuclear Safety in INFCIRC 872. The declaration does not establish new requirements but recommits the contracting parties to the implementation of the CNS principles and objectives to prevent accidents and mitigate radiological consequences, as discussed in Articles 6, 14, 17, 18, and 19. Section 2.4.1.2 of this report summarizes the United States' implementation of these CNS objectives.

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.

7.1 Legislative and Regulatory Framework

The Atomic Energy Act of 1954, as amended, provides the legal framework for all regulation of civilian nuclear installations. However, this act provides general principles and concepts and leaves to the regulatory body (now the NRC) to address the details through specific rules, regulations, or orders. The Energy Reorganization Act of 1974 abolished the Atomic Energy Commission and, in its place, created the NRC to regulate the safety and security of commercial nuclear activities and the U.S. Energy Research and Development Administration (ERDA) to continue Government-sponsored nuclear activities, including nuclear promotional activities. ERDA was subsequently incorporated into the U.S. DOE. The NRC implements the Atomic Energy Act though regulations that are issued in accordance with the Administrative Procedure Act, a law that provides general rules and procedures for all Federal agencies, including the NRC.

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

- The Treaty on the Non-Proliferation of Nuclear Weapons, ratified in 1970, provides the foundation for the U.S. commercial export controls.

- 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, mandates standards for the physical protection of 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 United States to report significant accidents to IAEA and any State affected by a transboundary radioactive release. The NRC would assist the U.S. Department of State in reporting significant accidents.

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

- The CNS, ratified in 1999, calls for periodic review meetings of all the contracting parties. Before the review meeting, each contracting party submits a National Report that details its commitment to nuclear safety. The NRC has the lead in preparing the National Report on behalf of the United States.

- 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 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, establishes a framework obligating the United States and other contracting parties to contribute to an international fund for compensation for "nuclear damage" resulting 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. In another example, in the wake of the Three Mile Island accident, President Carter directed the Federal Emergency Management Agency (FEMA) to direct all offsite emergency activities and review emergency plans in States with operating reactors. In a third 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 regulations and guidance.

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 a single licensing board member (i.e., presiding officer) or the Commission may also conduct hearings.

NRC licensing of nuclear power reactor facilities can take one of two approaches. The original licensing approach, under 10 CFR Part 50, requires two steps. In the first step, the NRC decides whether to grant a construction permit. In the second step, the NRC decides whether to grant an operating license once the plant has been constructed. The NRC licensed all current operating nuclear power plants in the United States according to this two-step process.

The alternative licensing approach, under 10 CFR Part 52, provides for combined construction and operating 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.

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.

An application for a combined license may (but is not required to) reference a standard nuclear reactor design that has been certified 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. A design certification is valid for 15 years and can be renewed for an additional 10 to 15 years.

The license for a nuclear power plant may be renewed for periods of 20 years. The NRC provides the licensing system for license renewal under 10 CFR Part 54.

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 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. Section 6.3.2 of this report discusses this further.

7.2.4 Enforcement

The Atomic Energy Act and the Energy Reorganization Act of 1974 provide the NRC with enforcement authority.

The Atomic Energy Act, Section 161, authorizes the NRC to conduct inspections and investigations and to issue orders necessary to protect public health and safety and to promote the common defense and security. Section 186 authorizes the NRC to revoke licenses under certain circumstances (e.g., for material false statements made to the agency, 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, or for a violation of the Atomic Energy Act or NRC regulation).

Various sections of Chapter 18 of the Atomic Energy Act also provide enforcement mechanisms for violation of NRC requirements. Section 234 authorizes the NRC to impose monetary civil penalties for violations of licensing requirements, not to exceed $100,000 per violation per day. However, that amount has been regularly adjusted for inflation since 1996. The NRC is currently required by the Federal Civil Penalties Inflation Adjustment Act of 2015 to adjust this maximum civil penalty amount annually. The amount is currently set at $298,211.

Section 232 authorizes the Attorney General, on behalf of the United States, to seek an injunction or other court order when, in the judgment of the Commission, any person has engaged or is about to engage in a violation of NRC requirements.

Section 223 of the Atomic Energy Act provides for varying levels of criminal penalties (i.e., monetary fines and imprisonment) for willful violations of the Atomic Energy Act, or of regulations or orders issued by the NRC under Sections 65, 161b, 161i, or 161o of the Atomic Energy Act. This section also allows the imposition of criminal penalties on certain individuals who are employed by firms constructing or supplying basic components of any utilization facility, including commercial nuclear power plants, 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 certain responsible persons at a firm constructing, owning, operating, or supplying components to a licensed or regulated facility for knowingly and consciously failing to provide the NRC with certain information relating to substantial safety hazards.

NRC regulations specify the procedures that the agency uses when exercising its enforcement authority against licensees or other persons subject to the NRC's jurisdiction. These regulations are found in 10 CFR Part 2, Subpart B, "Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties," which includes the following procedures:

- 10 CFR 2.201, "Notice of Violation," outlines the procedure for issuing a written notice of violation, including the content of the notice and explanation of any actions required by the recipient of the notice.

- 10 CFR 2.202, "Orders," explains the procedure for issuing orders, which may 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. A licensee must answer a demand for information. A person other than a licensee who is issued a demand for information 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 imposing civil penalties. The NRC initiates the civil penalty process by issuing a notice of violation and proposed imposition of a civil penalty. The agency gives the person charged with the civil penalty 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. The NRC gives a person charged with a civil penalty an opportunity to request 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.

Section 9.3 of this report discusses the NRC's enforcement process.

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.

8.1 The Regulatory Body

This section explains the NRC's mandate, authority and responsibilities, structure and position in the Government, and its financial and human resources, as well as its international responsibilities and activities, such as those related to international standards and IRRS and 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 authority 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 regarding 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, under the National Environmental Policy Act of 1969, as amended, the NRC conducts environmental reviews associated with its licensing responsibilities.

The NRC's licensing authority extends to other Government organizations (such as the Tennessee Valley Authority, which operates commercial nuclear power plants), but its 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 NRC is an independent regulatory agency within the executive branch of the Federal Government. The President cannot ordinarily direct the agency's regulatory decisions. The President can remove an NRC Commissioner only for cause—namely, "inefficiency, neglect of duty, or malfeasance in office." Congress cannot override the Commission's decisions, except by duly enacted legislation. Final Commission decisions can, however, be challenged in Federal appellate courts. When the Commissioners are acting in their adjudicatory role, they are bound to follow strict requirements, like those followed by Federal court judges, to ensure that persons outside the agency do not provide them with information that is relevant to the proceeding and is not made available to all parties.

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.

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 for 5-year terms. No more than three Commissioners can be a member of the same political party. 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. 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 and regional offices.

8.1.3.2 Component Offices of the Commission

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

- <u>Office of the Executive Director for Operations</u>. The Executive Director for Operations is the chief operating 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. The Executive Director for Operations is obligated to keep the Commission fully and currently informed of matters within its functions.

- 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 requires 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 directly 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 Commissioners, 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; responds promptly 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 is responsible for matters of law and legal policy, and provides 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 makes 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 directly to the Chairman and directs the agency's public affairs program, consulting with and advising agency officials while developing key communications strategies that support increased public confidence in NRC policies and activities. This includes keeping agency leadership informed on matters of public interest, influencing news coverage of the NRC's regulatory activities, and providing the public and the media timely, clear, and accurate information about NRC activities using a variety of communications vehicles, including 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.

- Office of Administration. The Office of Administration manages and provides centralized services in the areas of acquisition, property management, and administrative services, including support for agency directives, transportation, parking, translations, audiovisual needs, food services, mail distribution, labor services, furniture and supplies availability, NUREG publications, graphics, and printing services. The office develops policies and procedures and manages the operation and maintenance of NRC offices, facilities, and equipment. The office plans, develops, establishes, and administers policies, standards, and procedures for the overall NRC security program.

- Office of the Chief Human Capital Officer. The Office of the Chief Human Capital Officer provides overall management of the agency's human capital and human resources planning, policy, and program development. This includes overseeing the development and implementation of human resources management and information systems for staffing, SWP, and other corporate activities to support a dynamic workforce. The office implements NRC policies, programs, and services to provide for employment services and operations, training, employee and labor relations, organizational development, and workforce information and analysis, as well as administering and managing work life services programs, including oversight of the employee assistance program, child care facility, health unit, and fitness center. 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 Allegation Management Program. The office is responsible for external safety culture policy matters and partners with the Office of the Chief Human Capital Officer on the NRC's internal safety culture activities. The office oversees and manages the agency's Alternative Dispute Resolution Program, the Differing Professional Opinion Program, and the Non-Concurrence Process.

- 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 coordinates and oversees the development and update of agencywide information resources management policy. It manages the implementation of the Freedom of Information Act and oversees the agency's information collection activities.

- 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 investigation of all allegations of wrongdoing by other than NRC employees and contractors. It refers 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, 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; the safe decommissioning of materials, power and nonpower reactor sites, and the cleanup of contaminated sites; and safe uranium recovery activities. The office also leads, manages, and facilitates rulemaking activities in the agency.

- 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 provides technical advice, tools, and information to identify potential safety and security issues and resolve them as appropriate, assessing risk and other nuclear safety and security issues, and developing and coordinating regulatory guidance. This includes conducting confirmatory experiments and analyses, developing technical bases that inform NRC's safety decisions, and preparing NRC for the future by evaluating the safety aspects of new technologies and designs for nuclear reactors, materials, waste, and security. The office uses its own expertise and collaborates with partner offices at the NRC, commercial entities, national laboratories, other Federal agencies, universities, and international organizations.

- Office of Nuclear Security and Incident Response. The Office of Nuclear Security and Incident Response is responsible for accomplishing key components of the NRC's nuclear security program by evaluating and assessing technical issues involving security and emergency preparedness at domestic commercial nuclear facilities and for radioactive material in transit. The office conducts the agency's program for response to incidents. 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 enabling the agency to have a diverse and inclusive workforce, to advance equal employment opportunity for employees and applicants, to provide fair and impartial processing of discrimination complaints, to afford maximum practicable prime and subcontracting opportunities for small businesses, and to allow for meaningful and equal access to agency-conducted and financially-assisted programs and activities.

- 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 for U.S. licensed nuclear facilities. The regional offices also conduct oversight and inspection of decommissioning activities.

8.1.3.4 Advisory Committees

The NRC utilizes two advisory committees for the purpose of obtaining advice or recommendations: the Advisory Committee on Reactor Safeguards and the Advisory Committee on the Medical Uses of Isotopes. These committees are composed of experts in their respective fields, appointed from outside the agency. By law, committee meetings are presumptively open to the public.

- Advisory Committee on Reactor Safeguards. The Advisory Committee on Reactor Safeguards has statutory responsibilities as described in Section 29 of the Atomic Energy Act of 1954, as amended. The Committee reviews and advises the Commission on matters regarding the licensing and operation of production and utilization facilities, the adequacy of proposed reactor safety standards, technical and policy issues in the licensing of evolutionary and passive plant designs, specific generic matters, nuclear facility safety-related items, areas of health physics and radiation protection, and research activities, among others.

- 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.

In addition, although not an advisory committee, the NRC has a Committee to Review Generic Requirements, composed of NRC senior managers, that reviews proposed generic and facility-specific backfits that are to be imposed on all power reactors or selected nuclear materials facilities licensed by the NRC. The Committee to Review Generic Requirements ensures that proposed generic backfits 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 Section 191 of the Atomic Energy Act of 1954, as amended, Congress authorized the Commission to establish Atomic Safety and Licensing Boards and to appoint a panel of administrative judges from which board members may be selected. Members of this panel—either as a single presiding officer or in three-judge boards—conduct hearings for the Commission. Additionally, the panel performs 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, FEMA, U.S. Department of Homeland Security (DHS), U.S. Department of Justice, U.S. Department of Labor, U.S. Department of State, U.S. Department of Transportation, EPA, and Office of Management and Budget. Section 8.2 of this report discusses the NRC's relationship to DOE. The following summarizes the agency's relationships with the other identified components of the Federal Government:

- The White House Office. The White House cannot directly set NRC policy because of its status as an independent regulatory agency. However, in nonadjudicatory matters, such as rulemaking, the White House and executive branch officials may make their views known. The President nominates, subject to Senate confirmation, Commissioners and designates one Commissioner to serve as Chairman.

  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. 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 ensure that program activities are aligned with U.S. foreign policy objectives.

- Federal Emergency Management Agency (FEMA). FEMA assists the NRC's licensing process by conducting reviews and preparing findings and determinations on the adequacy of offsite radiological emergency plans and preparedness for NRC-licensed commercial nuclear power reactor facilities; and by presenting witnesses to testify at licensing hearings. FEMA also participates with the NRC in observing and evaluating offsite aspects of emergency exercises at nuclear plants. FEMA's findings are not binding on the NRC, but they support the NRC's overall determination of reasonable assurance and are presumed to be valid unless controverted by more persuasive evidence. FEMA is part of DHS.

- U.S. Department of Homeland Security (DHS). DHS is a cabinet department of the executive branch. Its mission is to secure the Nation from threats. The NRC routinely coordinates with DHS on infrastructure protection and cybersecurity issues.

- U.S. Department of Justice. 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, but the NRC has the right to appear and be represented by its own counsel. 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 asks 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 Organisation for Economic Co-operation and Development's NEA. 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. The NRC establishes requirements for the design and manufacture of packages for radioactive materials. U.S. Department of Transportation regulations cover shipments while they are in transit, including packaging, shipping and carrier responsibilities, and related documentation.

- U.S. Environmental Protection Agency (EPA). 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 recovery 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.

- Office of Management and Budget. The NRC submits its annual budget request to the Office of Management and Budget, which is an agency within the Executive Office of the President. The Office of Management and Budget assists the President in the preparation of the Federal budget, which is submitted to Congress for authorization, including funding for the NRC. In addition, the Office of Management and Budget approves the apportionment of current FY funding levels.

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.

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 discontinue its authority over most nuclear materials to those States willing to assume that authority within its borders. The NRC may not discontinue 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-eight 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 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 radiological health and radiological safety inspections.

The NRC works in cooperation with Federal, State, and local governments; interstate organizations; and federally recognized Tribes 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

Congress may pass legislation impacting nuclear safety or NRC operations. As noted above, the U.S. Senate also votes on whether to confirms the Presidents nominees to the Commission. Additionally 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 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 the Subcommittee on Environment and Climate Change have responsibility for 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 a variety of bilateral and multilateral activities related to statutory mandates, international treaties and conventions, cooperation and assistance, and research. U.S. law or international treaties and conventions mandate several NRC international activities; other activities are discretionary.

The NRC's international activities are integral to the NRC's public health and safety and common defense and security mission, as explained in the Commission's International Policy Statement, dated July 10, 2017 (79 FR 39415). They also support U.S. foreign policy for the safe and secure use of nuclear materials and for guarding against the spread of nuclear weapons. The agency actively implements 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 regulatory assistance to develop effective regulatory programs and implement rigorous safety and security standards. Some activities are carried out under the auspices of IAEA, NEA, or other international organizations. The NRC conducts other activities directly with counterparts 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 Treaty on Non-Proliferation of Nuclear Weapons, the Convention on Physical Protection of Nuclear Material, as amended, the CNS, the Convention on Early Notification of a Nuclear Accident, the Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency, the Convention on Supplementary Compensation for Nuclear Damage, and the 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 most of 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 participates in a wide variety of other 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 is statutorily mandated to serve as the U.S. licensing authority for exports and imports 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; 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 on export- or import-related matters that fall within these agencies' jurisdictions.

International Organizations and Associations. In consultation with 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 participates in a number of IAEA peer review missions. Some experts on these teams come from the NRC, while others come from industry. Examples of missions supported in 2017 and 2018 by the NRC or U.S. private industry include Emergency Preparedness Review, IRRS, International Physical Protection Advisory Service, OSART, and the Integrated Review Service for Radioactive Waste and Spent Fuel Management, Decommissioning and Remediation. On average, the NRC supports between 5–10 IAEA-sponsored peer review missions each year.

As discussed in Section 8.1.5.1 of this report, the NRC actively participates in the IAEA Commission on Safety Standards, all of the IAEA Safety Standards Committees, the IAEA Nuclear Security Guidance Committee, NEA Committees, and 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 has also continued its work both with IAEA and on a bilateral basis in support of countries seeking to enhance their nuclear regulatory programs. The NRC staff has been active in guidance document development and has participated in many workshops and training activities to provide "new entrant" countries with information and experience on building a robust, independent regulatory infrastructure. To that end, the NRC has participated actively in IAEA's Regulatory Cooperation Forum, with a senior NRC executive holding the position of Vice Chair since 2017.

Bilateral Relations. The NRC has arrangements to exchange technical information with nuclear safety agencies in 45 countries, Taiwan, and the European Atomic Energy Community. These arrangements establish the framework for the NRC's communications with foreign regulatory authorities regarding pertinent information with direct applicability to ensuring the safety and security of civilian uses of nuclear and radioactive materials globally. Activities under these arrangements include, but are not limited to: information exchanges on regulatory approaches and best practices; notification of potential safety concerns; accident and incident analyses at operating reactors; and cooperative research and code sharing programs. These arrangements also enable the NRC to provide training and health and safety assistance to countries as they work to develop their respective regulatory capabilities and nuclear safety infrastructure for oversight of a new nuclear power or radioactive materials program. In addition, the NRC engages with many countries either bilaterally or regionally on a limited basis where there is not yet a formal bilateral arrangement in place. NRC Commissioners travel internationally to share insights on a variety of topics with diverse technical and political counterparts. 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.

International Assistance Programs. Since the early 1990s, the NRC has continued to expand its international assistance program to countries developing or enhancing regulatory capacity for their 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 operating. Over the past decade, the NRC's reactor-related assistance programs expanded to include a focus on new reactor issues, aging management, and physical protection. The NRC provides technical expertise, training, and technology neutral information covering a broad range of topics relevant to organizational infrastructure and regulatory programs relating to nuclear power programs. The NRC's International Regulatory Development Partnership program provides training to regulatory bodies in nuclear orientation (codes and standards, fundamentals of reactor regulation and safety, PRA, and quality assurance), agency infrastructure development (organizational planning 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).

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 meet mutual research needs with greater efficiency.

Taken together, the suite of international activities—treaty implementation, export-import licensing and bilateral and multilateral cooperation and assistance—facilitate the NRC's strategic goal to support U.S. interests in the safe and secure use of nuclear materials and in nuclear nonproliferation.

8.1.5.1 International Standards

The NRC, along with several other U.S. Federal agencies, actively participates in the development of IAEA's safety standards. Where appropriate, the NRC also references the safety standards in NRC regulations and regulatory guidance.

NRC senior management and staff represent the agency at the IAEA Commission on Safety Standards and the IAEA Safety Standards Review Committees. Additionally, NRC senior technical experts support the development of the safety standards by providing 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 safety standards are used as reference documents to inform the development of requirements and guidance in the NRC's reactor, radiation protection, transportation, waste management, and emergency preparedness and response programs.

Differences in the application of IAEA safety standards and NRC regulations largely stem from the fact that NRC regulatory infrastructure predates most IAEA safety standards. Furthermore, NRC requirements are written with a greater level of detail than the IAEA safety standards. Despite these differences, the NRC agreed with recommendations from the 2010 U.S. 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 it updates agency regulations and guidance documents. The NRC's policy guidance directs the staff to consider IAEA standards as a point of reference when drafting or revising RGs, and to consider direct endorsement of the IAEA standards when appropriate. In the past 2 years, the NRC has published 16 new or revised RGs that harmonize with or reference IAEA safety standards and the completion of an additional nine RGs is anticipated by summer of 2020.

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 two 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 IRRS followup mission closed one of the two recommendations and 19 of the 20 suggestions. One new suggestion was opened concerning the transition of operating reactor plants to decommissioning. The followup mission also reviewed the NRC's response to the Fukushima accident. 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 on the NRC's public Web site.

The NRC 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 on the 2010 and 2014 IRRS missions. The letter, which is available on the NRC's public Web site, gives the final response to all items.

8.1.5.3 Operational Safety Assessment Review Teams

The NRC coordinates with INPO to facilitate 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. To support and encourage this international program, the NRC licensees that host OSART missions can receive some reduced NRC inspections under the Reactor Oversight Process based on which technical areas the OSART team reviews.

In August 2017, Sequoyah Nuclear Power Plant, located in Tennessee, hosted an OSART mission. The team identified six recommendations, 13 suggestions, and two good practices. The results of the OSART are documented in the 2018 IAEA-NSNI/OSART/195/2017, "Report of the Operational Safety Review Team (OSART) Mission to the Sequoyah Nuclear Power Plant 14-31 August 2017," which is available on the NRC's public Web site. Sequoyah's management expressed its commitment to addressing the issues identified and hosted a follow-up visit in April 2019.

The next OSART in the United States is tentatively scheduled to take place in 2020.

8.1.6 Financial and Human Resources

8.1.6.1 Financial Resources

As of October 1, 2018, 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 2019 enacted budget was $911 million, including the budget for the Office of the Inspector General.

The NRC FY 2019 budget was financed with $780.8 million from user fees and $130.1 million from the U.S. Government's General Fund.

8.1.6.2 Human Resources

The NRC uses SWP to maintain its comprehensive human capital management system. SWP is a structured and data driven process, to develop short-term and long-term strategies that enable the NRC to recruit, retain, and develop a skilled and diverse workforce with the competencies and agility to address emerging needs and workload fluctuations. In 2016, the NRC began to implement enhancements in how it plans and maintains its workforce to better accomplish its nuclear safety and security mission. In addition to SWP, the enhancements are intended to improve the effectiveness, efficiency, agility, consistency, and standardization of the process.

SWP improves workforce management by anticipating and planning for changes in industry, constraints on the budget, and many other internal or external factors. By strategically managing its workforce, the NRC will be able to reduce surpluses or shortfalls in each of the skill sets needed, determine the workforce size, and build an agile workforce that enables the agency to shift qualified employees or their work assignments to meet the demands of a changing environment with speed and flexibility. Employees are empowered to use this information to plan their personal and career development with a greater understanding of the agency's short-term and long-term workforce plans.

The SWP process takes place on an annual cycle and it integrates with existing agency processes: strategic planning, staffing, budget formulation, performance management, and training and development. The process has six defined steps:

(1) Set Strategic Direction. The effort uses the Strategic Plan and the Agency Environmental Scan to monitor internal and external opportunities and risks that may influence current and future workloads. The results of the workload forecast provide strategic insights into workforce needs, as well as potential changes to positions or the NRC's structure.

(2) Workforce Demand Analysis. The analysis uses the workload forecast to determine the core positions needed to perform the work, the number of people in each core position, and the proficiency levels needed now and in the future, including competencies required to meet emerging needs.

(3) Workforce Supply Analysis. The analysis reviews the current workforce and forecasts the number of employees and associated competencies into the future. The analysis considers attrition risk, position risk, and skill level for each employee.

(4) Gap Analysis and Risk Assessment. The effort determines and prioritizes the gaps and surpluses that may exist between the information collected from steps 2 and 3. The results highlight the associated risks.

(5) Develop and Execute Strategies. Short-term and long-term strategies and action plan(s) are developed to address anticipated surpluses or gaps in the workforce.

(6) Monitor, Evaluate, and Revise. Strategies are continuously monitored, evaluated, and revised to make course corrections and to address new workforce issues and changes in internal and external environments.

Recruitment and Hiring Process. Several internal and external factors are driving changes in hiring practices at the NRC, including flat or decreasing agency budgets and lower than projected numbers of new reactors. While near-term hiring will center on the most critical skill sets, the NRC will continue to emphasize Governmentwide programs, such as hiring of the disabled, employment of veterans, enhancing diversity, and supporting the agency's Comprehensive Diversity Management Plan.

The NRC continues to use 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 with the critical skills needed to perform mission-related work. 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 staff members to work remotely and reduce their commute times.

Work Environment. The NRC regularly solicits feedback to gain independent and diverse perspectives on ways to improve the agency's work environment. In view of that, the agency often explores various channels to seek meaningful insights about employees and their work experience. One such mechanism is 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. Conducted annually, the Federal Employee Viewpoint Survey is mandated by the Office of Personnel Management's regulations. The Safety Culture and Climate Survey is administered by the NRC's Office of the Inspector General approximately every 3 years. These surveys provide unique, but also overlapping, insights into the NRC workplace that together build a comprehensive picture of employees' experiences with their job, supervisors, and work units. Both surveys have consistently revealed that the NRC is a top performing organization within the public sector and ranks competitively against private sector benchmarks. The agency focuses action planning on areas identified in both surveys, along with reinforcing the existence of a positive environment for raising concerns and valuing human differences. Section 10.3.3 of this report discusses the NRC's safety culture in more detail.

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 fosters continuing improvement in performance through knowledge management, performance feedback, training, coaching, and mentoring.

The NRC continues to implement blended learning strategies that combine educational techniques to optimize knowledge transfer. 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 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 that the NRC captures and preserves 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 in the success of the Knowledge Management Program is the system of governance provided by 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 to ensure that the 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. For example, 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.

8.1.7 Openness and Transparency

The NRC established openness as one of five Principles of Good Regulation in 1977 to guide 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.

The NRC extends opportunities to participate in the agency's regulatory process to the public, Congress, other Federal agencies, States, local governmental bodies, Indian Tribes, industry, technical societies, the international community, and citizen groups. Many NRC programs and processes provide the public with access to NRC staff and other resources; seek to make communication with stakeholders clearer and more accurate, reliable, objective, and timely; and help to ensure that the reporting of nuclear power plants' performance is open and objective.

<u>Access to NRC Documents</u>. 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's skilled technical and reference librarians provide information and research assistance directly to stakeholders, including environmental groups, licensees, the legal community, and concerned citizens.

To ensure that the public has access to the information it needs, the NRC makes documents available to the public, unless there is a specific reason for information to be withheld. 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 excludes security-related, proprietary, or other sensitive information. In 2018, approximately 67,000 public users accessed ADAMS more than 337,000 times and requested documents more than 16.2 million times.

The NRC reports to Congress each year on 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 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 NRC's Open Government Plan, last updated in 2018, describes concrete, measurable steps the agency has implemented to openly conduct its work and publish information online. The plan covers 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.

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 Web 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 shares information with stakeholders and the public on its public Web site. In 2018, the NRC's Web site had more than 2.2 million individual visitors. The Web site was visited more than 4.9 million times, and visitors viewed more than 31.3 million pages. The site provides information on Commission decisions, hearing transcripts, inspection reports, enforcement actions, licensing reviews, petitions, event reports, and daily plant status. It includes a tool to locate information on facilities the NRC regulates and details on U.S. nuclear power plant performance. It also contains considerable 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 on topics such as license applications, major licensing decisions, enforcement actions, major public meetings, opportunities for hearings, and other avenues for public involvement. Users may sign up through the Web site to receive automatically several types of documents, including press releases, 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 NRC video streams many Commission meetings over the Internet. More recently, the agency expanded Web casting to other high-interest meetings, conferences, and adjudicatory hearings. These Web casts are available for viewing live and are archived for viewing later. The agency also uses webinars to more effectively share information and communicate with the public.

Social Media. The NRC embraces social media as an important tool for reaching a broader public audience. The agency uses these social media platforms to give the public information, raise awareness, explain technical activities, and spotlight accomplishments. The NRC's Office of Public Affairs manages these tools, but NRC staff members at all levels help ensure that the agency is meeting the communication needs of all its offices, both at headquarters and in the regions.

The NRC's social media platforms have been integrated into the agency's crisis communication strategy. Agency personnel regularly simulate external communications using social media during exercises. The NRC's Facebook, Twitter and YouTube platforms have been used effectively in real-life situations such as severe weather events to communicate timely and relevant information.

The NRC uses its Twitter account, launched in August 2011, to alert the public to new press releases, *Federal Register* notices, licensing decisions, guidance documents, important personnel changes, and any topic that might emerge. The NRC has live tweeted from high-profile meetings, including the annual Regulatory Information Conference. As of July 2019, the NRC had more than 10,600 Twitter followers. The agency sent a total of 3,822 tweets over 97 months for an average of 39 per month.

The agency launched its Facebook page in August 2014. Since that time, its page has gained more than 6,900 likes and 225,700 engagements on 990 posts. The NRC uses Facebook to inform the public about specific regulatory activities, to underscore national and agency events, to highlight employee accomplishments, and to educate and inform its audience about nuclear and regulatory topics.

The NRC's YouTube channel and Flickr photo gallery provide 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 NRC staff activities. These forums visually document the agency's work and introduce the people who carry out the agency's mission. Since launching the YouTube channel in August 2011, the agency has posted about 200 videos, which have received nearly 322,900 views. More than 1,650 users subscribe to the NRC YouTube channel and are notified each time new content is posted. Since February 2012, the NRC has published about 2,760 photos and graphics to its Flickr account which have been collectively viewed nearly 1.3 million times.

Public Meetings. The public may participate in a variety of ways before the NRC issues certain licensing actions. To ensure this involvement is meaningful, the NRC actively communicates with stakeholders on how the NRC makes decisions—including the agency's role, processes, and activities. The NRC meets 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's use of Web conferencing allows 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 identify ways the NRC can improve public meetings.

The NRC staff hosts and participates in 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 allows 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 live streams key conference events.

Plain Language. 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.

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

U.S. law, through legislation enacted by the U.S. Congress, ensures the effective separation of 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 in 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 U.S. Office of Management and Budget 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 “Department of Energy Organization Act” in 1977 subsequently brought several 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 issued by the U.S. Office of Government Ethics. These rules state principles of ethical conduct and are intended to ensure that every citizen can have confidence in the integrity of the Federal Government. The rules create standards and obligations that Federal employees must follow in situations that pose conflicts of interest or may raise ethics concerns. For example, the rules restrict an employee’s ability to accept gifts from regulated entities; prohibit an employee from participating in matters that would affect the employee’s personal financial interests; provide standards for recusal in matters involving persons with whom the employee has certain personal or business relationships, such as a matter involving a family member or recent former employer; and preclude the employee from using a public position for private gain.

In addition to these Governmentwide rules, the NRC has issued two supplementary ethics rules that apply to its employees. First, the NRC has established 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 issued by any company appearing on the Prohibited Securities List. Second, the NRC has a rule that requires employees to obtain prior approval before engaging in any compensated outside employment with certain types of employers, including any organization directly engaged in activities in the commercial nuclear field.

When an NRC employee leaves the agency for a non-Federal employer, the employee must also comply with certain postemployment rules that restrict the former employee’s ability to attempt to influence the Federal Government on behalf of his or her non-Federal employer. The scope and length of the restriction depends on the former employee’s position at the time he or she leaves the NRC and the extent of the employee’s previous participation in the matter on which he or she seeks to represent the non-Federal party.

In addition to these rules, since 2009 all full time political appointees in the executive branch, including at the NRC, have been required by the President to sign an Ethics Pledge as a condition of their appointment. The current Ethics Pledge, contained in Executive Order 13770, “Ethics Commitments by Executive Branch Appointees,” dated January 28, 2017, further limits an appointee’s ability to accept gifts from lobbyists, restricts or prohibits certain lobbying activities after leaving Government, and imposes other 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. 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, the Petition for Enforcement Process, and the Allegation 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, grants the NRC authority to issue licenses for production and utilization facilities for commercial or industrial purposes, which include 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 of 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.

If the Commission determines that the licensee is not complying with its license or the Commission's rules or regulations, the NRC takes appropriate action to ensure that the facility returns to compliance. Sections 7.2.4 and 9.3 of this report provide more details about the NRC's Enforcement Program. Section 6.3.2 of this report discusses the NRC's Reactor Oversight Process.

9.3 Mechanisms To Enforce the Licensee's Responsibility to Maintain Safety

9.3.1 Enforcement Program

As discussed in Article 7, the NRC has enforcement powers. As discussed in Sections 7.2.3 and 7.2.4, the Reactor Oversight Process complements, and works in conjunction with, the Enforcement Program. 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 May 15, 2018. Because it is a policy statement and not a regulation, the Commission may deviate from it, as appropriate for the circumstances of a particular case.

The NRC has three primary enforcement sanctions available:[8]

(1) Notices of Violation. A notice of violation identifies a requirement and how it was violated, requires corrective action, and normally requires a written response.

(2) Civil Penalties. A civil penalty is a monetary fine used to emphasize compliance to deter future violations and to focus compliance on significant violations.

(3) Orders. Orders can be used to modify, suspend, or revoke licenses, or they may require specific actions by licensees or persons. Orders extend to any area of licensed activity that affects public health and safety or the common defense and security. The NRC issues notices of violations and civil penalties based on 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 actual and potential safety consequences; the potential for impacting the NRC's ability to perform its regulatory function; and any willful aspects of the violation. Based on the significance of the violation, the NRC assigns a severity level, ranging from Severity Level IV (more than minor concern) to Severity Level I (the most significant). Findings and associated violations assessed through the Reactor Oversight Process significance determination process (described in Article 6) are assigned the colors green, white, yellow, and red based on increasing risk significance.

8 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 on 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—no civil penalty, a base civil penalty, or twice the base civil penalty. A base civil penalty has been established in the NRC Enforcement Policy 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. After 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 list of public meetings on the NRC's public Web site (https://www.nrc.gov/pmns/mtg). 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 on the NRC's public Web site (https://www.nrc.gov/about-nrc/regulatory/enforcement/current.html).

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

Table 3 Recent Enforcement Actions

	Calendar Year		
	2016	**2017**	**2018**
Notices of violation without civil penalties	17	22	7
Civil penalties	1	0	3
Orders without civil penalties	5	3	2
Total enforcement actions	**23**	**25**	**12**

9.3.2 NRC Petition for Enforcement Process

Among the agency tools established for the public, industry, and NRC employees to raise safety concerns, the NRC's petition process described in 10 CFR 2.206, "Requests for Action Under This Subpart," allows any person to raise potential health and safety concerns and ask the agency to take specific enforcement actions against an NRC licensee or licensed activity.

Most aspects of the 10 CFR 2.206 petition process are public including meetings with the petitioner and petition related documents. The NRC's procedures governing this petition process emphasize timely responses 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 NRC staff personally and comment on the agency's decision.

The NRC's review of a 10 CFR 2.206 petition may include the formation of a Petition Review Board made up of cognizant NRC staff and managers. The board can assess the potential issue and rule on the requested enforcement action. If warranted, the Commission may ultimately grant a request for action, in whole or in part, take other action that satisfies the concerns raised by the requester, or deny the request. If a request is granted, the NRC may modify, suspend, or revoke a license, or take other appropriate enforcement action, to resolve the problem(s) identified in the petition.

9.3.3 Allegation Program

As a part of the overall safety culture expectations, 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 are encouraged to raise safety concerns, are free to raise concerns to both their management and NRC without fear of retaliation, where concerns are promptly reviewed, given the appropriate priority, and are appropriately resolved, and where timely feedback is provided. These expectations are communicated through the NRC's "Freedom of Employees in the Nuclear Industry To Raise Safety Concerns Without Fear of Retaliation Policy Statement" (61 FR 24336; May 14, 1996), safety conscious work environment guidance documents, and other related regulatory tools. Section 10.3 of this report discussed the NRC's safety culture principles and objectives in more detail.

Additionally, workers and members of the public may bring their concerns about safety or regulatory issues directly to the NRC. The NRC documents, evaluates, and assesses the validity and safety significance of these concerns by using the guidance in MD 8.8, "Management of Allegations," dated January 29, 2016. The Allegation Program's primary purpose is to provide an alternative method for individuals to raise safety or regulatory issues. The agency maintains a toll-free safety hotline and e-mail account for reporting such concerns. NRC management, staff, and inspectors, including the resident inspectors at nuclear power plant sites, are trained and available to receive such concerns.

Historically, industry workers or members of the public report approximately 400 potential allegations directly to the NRC's Allegation Program each year. 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. Additional information about the Allegation Program, including frequently asked questions, trends, and statistics, can be found on the NRC's public Web site: https://www.nrc.gov/about-nrc/regulatory/allegations-resp.html.

9.4 Openness and Transparency

U.S. nuclear power plant licensees are required to demonstrate that the appropriate governmental authorities have the capability (e.g., sirens, tone alert radios, and route alerting) to alert the public of a nuclear power plant emergency event 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 Article 16 of this report. Licensees also provide 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. 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, as well as traditional and social media.

Each licensee has established a joint information center, which 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.

Section 8.1.7 of this report describes the NRC’s openness and transparency objectives.

9.5 Financial and Human Resources

9.5.1 Financial Resources

Licensees have financial responsibilities in the event of an accident. Section 182.1 of the Atomic Energy Act, as amended, provides the basis for the NRC’s onsite property damage insurance requirements for operating nuclear power reactors 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 case 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 in the form of liability insurance for claims arising from accidents. Sections 11.1.3 and 11.1.4 of this report provide additional information on liability insurance.

9.5.2 Human Resources

This responsibility for safety is addressed, in part, by having trained and qualified operators. In 10 CFR 50.54, “Conditions of Licenses,” the NRC 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, “Training and Qualification of Nuclear Power Plant Personnel,” 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 safely operate and maintain the facility 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 Section 11.2 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. Part 3 of this report presents additional information on licensee training and accreditation programs.

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.

The NRC's mission is founded on nuclear and radiological safety and regulatory activities pertaining to nuclear installations reflect the risk-informed, performance-based approach that the NRC takes to fulfilling its mission. The NRC has several policy statements in place that describe the Commission's perspective on nuclear safety (e.g., 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 to achieve nuclear safety at nuclear installations.

10.1 Background

The NRC has a longstanding goal of moving 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. In a risk-informed approach, 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 address only a few design basis conditions and would rely 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 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 gives the licensee more flexibility in meeting the design or process objective. Implemented together, the risk-informed and performance-based approaches use risk insights, engineering analyses, judgment, the principles of defense-in-depth and safety margins, and performance history to achieve the following:

- 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 rewards 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, as described in Section 2.3.2.8 of this report.

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.

10.2.1 Applications of Probabilistic Risk Assessment

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 the agency has used risk information to evaluate proposed changes to the current licensing bases for individual plants. For example, as described in Section 2.3.2.8 of this report, 10 CFR 50.69 allows licensees to use a risk-informed approach to categorize SSCs and assign special treatment requirements, according to their safety significance. As another example, as described in Section 6.3.8 of this report, 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.

The NRC conducts research and collaborates with organizations that develop consensus standards to improve data and methods used in risk analysis. For example, the NRC worked with ASME and ANS to update the national consensus standard for PRA quality, ASME RA-Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," which the NRC later endorsed in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," Revision 2, dated March 2009.

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.

For new reactors licensed under 10 CFR Part 52, the NRC requires applicants to describe the design-specific PRA and its results for a design certification application and a plant-specific PRA and its results for a combined license application. In addition, the NRC requires the holder of a combined license to develop a Level 1 and a Level 2 PRA before initial fuel load. 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 before 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.

10.2.2 Level 3 Probabilistic Risk Assessment Project

As directed in 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, SFPs, and dry cask storage.

The full-scope site Level 3 PRA project has 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 the completion of NUREG-1150, “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 advances 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, a two-unit PWR site was selected as the reference site for the Level 3 PRA study. To enhance the study’s efficiency, the Level 3 PRA project team is leveraging information from approximately the year 2012 on the reference site, the associated reference site’s PRAs, and related research efforts.

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

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

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. An annual update on the status of the Level 3 PRA study can be found on the NRC’s Web site at https://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp/reactor-safety-operating.html#level-3
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10.3 Safety Culture

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

10.3.1 Safety Culture Policy Statement

Operating experience has shown the value of establishing and maintaining a positive safety culture. The NRC's "Final Safety Culture Policy Statement," dated June 24, 2011, 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 to identify discrepancies that might result in error or inappropriate action

After publication of the policy statement, the NRC engaged the INPO, 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 greater clarity and 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.

10.3.2 NRC Monitoring of Licensee Safety Culture

10.3.2.1 Background

Section 6.3.2 of this report describes the Reactor Oversight Process. Based on lessons learned from the reactor pressure vessel head degradation event at Davis-Besse Nuclear Power Station 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 degradation of plant performance.

10.3.2.2 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 options to address performance, as described below.

The Reactor Oversight Process uses a graded approach, such that plants that are performing in a specified manner warrant a routine level of inspection and oversight. However, as licensee performance deteriorates, inspection and oversight increase to ensure safe plant operation. The Reactor Oversight Process continues 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 25, 2019. For most licensees (i.e., those 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, NRC inspectors have the option to review licensee self-assessments of safety culture. This IP also instructs NRC 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, “Follow-up 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 of 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.

If licensee performance declines (Regulatory Response column, Column 2, of the Reactor Oversight Process Action Matrix), the NRC inspectors, through a specific supplemental IP, verify that the licensee's causal evaluation, 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 causal evaluation for the risk-significant finding(s) will determine whether any safety culture attribute contributed to the risk-significant performance issues. If, through the performance of a supplemental inspection using IP 95002, "Supplemental 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 ask 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 expects that the licensee will 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," dated December 2015, to provide guidance for these assessments.

Consideration of safety culture within the Reactor Oversight Process provides 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 that 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.

10.3.3 The NRC Safety Culture

The NRC fosters a culture in which all employees are encouraged to exemplify 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 demonstrates the agency's collective commitment to emphasize safety as the priority in its 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 that it remains a model regulator.

The NRC acknowledges that 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. Although many similarities in 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 his or her behaviors and attitudes that 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 its provision of 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 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 Opinion Program," dated August 11, 2015. 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 Opinion Program also support the NRC's openness value, in that when the process is complete, an employee can ask that the records be made public.

(3) The NRC conducts assessments of its 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 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 like this makes it possible to compare results over time to assess 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.

10.4 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 safety and security activities be integrated 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 focuses on 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, training and inspection to recognize, establish, and improve this interface. For example, the NRC has been working multilaterally with the IAEA and bilaterally with its international counterparts to promote this concept. In March 2009, the NRC issued 10 CFR 73.58, "Safety/Security Interface Requirements for Nuclear Power Reactors," which requires licensees to assess and manage changes to safety and security activities. In April 2015, the NRC issued Revision 1 to RG 5.74 adding cybersecurity to the safety and security assessment required in 10 CFR 73.58.

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 that licensees are complying with the regulations, the NRC has incorporated the evaluation of the licensee's safety and security interface processes into its IPs. Section 6.3.2 of this report discusses the Reactor Oversight Process in more detail.

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 for financial resources that licensees must have to support the nuclear installation throughout its life, and the regulatory requirements for qualifying, training, and retraining personnel.

11.1 Financial Resources

Currently, the NRC financial qualification regulations are codified in 10 CFR Part 50 and 10 CFR Part 52. 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 demonstrate 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 staff has determined that the details of these arrangements may go beyond the NRC's mandate of ensuring public health and safety. Therefore, the Commission is considering the staff's recommendation on whether to conform 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 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 property insurance and offsite liability insurance. This insurance provides the licensee with financial protection for any claims of bodily injury and property damage resulting from a nuclear incident and helps pay onsite recovery costs. Sections 11.1.3 and 11.1.4 of this report provide additional information.

The NRC also maintains decommissioning funding and related reporting requirements under 10 CFR 50.75 and 10 CFR 50.82 throughout the life of a reactor facility, and regularly reviews the status of licensees' decommissioning trust funds. These detailed reviews provide the NRC with 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, and license transfers.

Section 182.a of the Atomic Energy Act, as amended, states the following:

> 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 a lesser standard as in 10 CFR Part 70. The rulemaking would permit the inclusion of a license condition(s) to demonstrate that an applicant appears to be financially qualified and require the applicant to submit a plan for how it will proceed to finance the construction and operation of the facility. This plan, referred to as an Applicant Financial Capacity Plan, would ensure that the applicant has a well-articulated understanding of the size and scope of the project it is undertaking and the financial capacity to obtain the necessary financing when ready to start construction. The NRC staff provided the proposed rule to the Commission, and anticipates that the rulemaking activities, including opportunities for public involvement, will continue through calendar year 2019.

11.1.1.1 Construction Permit Reviews

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

NUREG-1577 Revision 1, provides staff with 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.

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, the NRC has broad authority under the Atomic Energy Act and NRC regulations in 10 CFR 50.54(cc), 10 CFR 50.54(f), and 10 CFR 2.102, "Administrative Review of Application," to obtain information from its licensees and applicants, as necessary, to protect public health and safety.

11.1.1.3 Combined License Application Reviews

As authorized in 10 CFR Part 52, applicants may apply for a combined construction permit and operating license. Under 10 CFR 52.77, "Contents of Applications; General Information," such applications must contain all of the information required under 10 CFR 50.33, "Contents of Applications; General Information," including information about financial qualifications. 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 Reviews of License Transfers

The provisions 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.

For an applicant seeking the transfer of a license of a decommissioning plant, an applicant's financial qualifications for decommissioning would be reflected in information that it submits to show that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated costs for decommissioning the facility and managing irradiated fuel.

NUREG-1577 provides staff with 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 provisions at 10 CFR 50.80(b) require license transfer applicants to include information with respect to, among other things, the financial qualifications of the proposed holder of the license as required in 10 CFR 50.33(f). In the case of license transfers, NUREG-1577 requests staff to: (1) determine whether the proposed holder of the license will remain an electric utility following the direct or indirect transfer; (2) for nonelectric-utility applicants, review 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, evaluate the ownership or participation agreement with its owners or other responsible party, and (3) identify all parent companies that are not licensed by the NRC or did not undergo a 10 CFR 50.80 review.

11.1.2 Financial Assurance 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 include, 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 as to rates of escalation in decommissioning costs, rates of earnings on decommissioning funds, and rates of other factors used in funding projections
- any contracts on 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 annually under 10 CFR 50.82. They include information on 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 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, and termination of the NRC-issued license.

In a rulemaking, the NRC staff is proposing to amend its decommissioning regulations to provide an appropriate regulatory framework for production and utilization facilities transitioning from operations to decommissioning. The draft proposed rule written by the NRC staff would amend NRC's regulations to allow the use of decommissioning funds collected and kept in an external trust, during decommissioning, for spent fuel management and for decommissioning of specific license independent spent fuel storage installations, if the licensees meet certain conditions. Additionally, the draft proposed rule would modify decommissioning funding reporting requirements, clarify decommissioning funding assurance requirements, and eliminate any duplicative regulations. Section 2.3.3.3 of this report provides additional information about this rulemaking.

11.1.3 Financial Protection Program for Liability Claims Arising from Nuclear Incidents

The Price-Anderson Act of 1957, which was codified in Section 170 of the Atomic Energy Act, as amended, governs the U.S. financial protection program. Along with related definitions in Section 11, Section 170 provides 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 funds are available to the public for liability claims if such an incident were to occur.

Congress most recently revised the Price-Anderson Act in 2005, when it renewed the insurance requirements for nuclear facilities until 2025. Under the current law, power reactors are subject to a multilayered financial protection framework. Power reactors that are 100,000 kilowatts electric or more must maintain the maximum amount of private liability insurance available to the industry, currently $450 million, and contribute to a secondary funding pool that is enacted only if the primary layer of financial protection is exhausted. The NRC is required to adjust the amount of secondary financial protection for inflation every 5 years based on the aggregate change in consumer price index.

As noted above, reactor operators must pay into a funding pool for the secondary layer of financial protection, called the “retrospective premium pool,” in maximum annual installments not to exceed $20.496 million, up to a total of $131.056 million for each reactor. These payments are required if a nuclear incident exhausts the first layer of financial protection, currently $450 million, and only if additional funds are needed to pay the damages. Upon petition to a U.S. district court, if the court determines that public liability may exceed the maximum amount of financial protection available from the primary and secondary layers, each licensee would be assessed a pro rata share of this excess not to exceed 5 percent of the maximum deferred premium ($131.056 million)—approximately $6.553 million per reactor. With 99 reactors[9] currently participating in the system, the total financial protection available under the Price-Anderson Act for any one incident is approximately $14 billion (i.e., $450 million of primary coverage for the site plus the secondary pool ($131.056 million per reactor times 99 reactors) plus $6.553 million per reactor times 99 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. If the second tier is depleted, Congress will determine whether additional disaster relief is required to protect public health and safety.

9 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. As of August 2019, in the United States, there are 97 nuclear reactors currently in operation.

The public benefits significantly from another feature of the Price-Anderson Act. All economic liability is channeled to the operator, which makes proof of fault unnecessary for payment of a claim. This feature was intended to help ensure that potential claims are resolved as expeditiously as possible 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 paid out a total of approximately $71 million in claims and litigation costs in association with the Three Mile Island incident in 1979.

The NRC staff is proposing to amend its financial protection regulations under 10 CFR Part 140 and 10 CFR 50.54(w) to address instances when a decommissioning reactor licensee may not need to maintain its full amounts of offsite liability insurance and onsite property insurance. Reductions in insurance amounts may be warranted commensurate with the reduction in risk at a reactor in decommissioning. Section 2.3.3.3 of this report presents additional information about this rulemaking.

Separate from the Price-Anderson Act, the United States is a party to the Convention on Supplementary Compensation for Nuclear Damage, which was developed under the auspices of IAEA to be the basis for a global nuclear liability regime. Section 8.1.5 of this report lists treaties that legally bind the U.S. Government's peaceful uses of nuclear energy and nuclear applications.

11.1.4 Insurance Program for Onsite Property Damages Arising from Nuclear Incidents

Among other sections of the Atomic Energy Act, Section 182.a gives the basis for the NRC's onsite property damage insurance requirements in 10 CFR 50.54(w) for operating nuclear power reactors. Onsite insurance provides the licensee with financial protection to stabilize and decontaminate the reactor and reactor site at which the reactor experiencing an incident is located.

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, 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 done on a control room simulator. Although the NRC does not mandate specific simulator training requirements (i.e., each facility licensee determines simulator training 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, heatups, cooldowns, 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, periodic emergency response training is conducted in the simulator using short (approximately 1-2 hour) scenarios. The licensee; and State and local emergency response organizations are assessed once every 2 years using scenarios lasting several hours during an exercise observed by the NRC and FEMA.

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 11, issued in February 2017. 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, 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, dated 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 consistent with 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 means 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. Part 3 of this report provides additional information about the INPO accreditation process.

The NRC monitors industry performance in implementing the training requirements of 10 CFR Part 50 and 10 CFR Part 55 by (1) reviewing LERs and inspection reports for training issues, (2) observing the accreditation process, and (3) reviewing the results of operator licensing activities. IP 71111.11, "Licensed Operator Requalification Program and Licensed Operator Performance," dated September 24, 2014, gives guidance for periodically inspecting the licensed operator requalification training program at every facility. 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, dated 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 2016, there has been no notable increase in the trends associated with training deficiencies and operator errors.

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 discusses human factors regulatory review and control activities of items such as plant design and modifications, organizational issues, staffing, and fitness for duty. This section also explains how human factors activities are integrated in the Reactor Oversight Process and how feedback and experience in human factors is considered in the regulatory program.

12.1 Overview of Regulatory Requirements

People are integral to the safe operation of a nuclear power plant. In recognition of this, following the Three Mile Island accident, the NRC began focusing on protecting the people that form the plant staff and ensuring that they have adequate training to perform their assigned tasks. The NRC also began studying factors affecting performance, such as the effects of shift work on health and the potential benefits of control room simulators to training.

Currently, the NRC conducts a range of activities in the areas of human and organizational factors to ensure that human performance is properly addressed using a risk-informed and performance-based regulatory framework. These activities include reviews of licensee submittals, inspections of licensee facilities and activities, and analyses of industry performance. Through these activities, the NRC addresses human performance from multiple perspectives, including human factors engineering, organizational factors, human performance, worker fitness for duty, and human reliability analysis.

12.2 Regulatory Review and Control Activities

12.2.1 Nuclear Power Plant Design and Modifications and Operator Actions

The NRC evaluates the human factors engineering design of the main control room and some control centers outside of the main control room using NUREG-0800, Chapter 18, “Human Factors Engineering,” Revision 3, dated December 2016; NUREG-0700, “Human System Interface Design Review Guidelines,” Revision 2, dated May 2002; and NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 3, dated November 2012. These documents provide guidance for the review of human-system interface issues. The NRC also uses NUREG-1764, “Guidance for the Review of Changes to Human Actions,” Revision 1, dated September 2007, to review license amendment requests that credit the use of manual actions.

Additionally, IN 97-78, “Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times,” dated October 23, 1997, identifies references that the NRC uses to review the completion times of operator manual actions and how the actions will be reflected in the licensee’s emergency procedures and operator training. In October 2007, the staff published NUREG-1852, “Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire,” for use in evaluating exemptions from fire protection requirements that assume credit for timely manual actions. Methods described in NUREG-1852 have also been successfully used to credit operator actions not related to fire.

The NRC reviews license amendment requests for operating plants that involve aspects of human and organizational factors. Examples include crediting operator manual actions in amendments to plant technical specifications and increasing the reactor's authorized power level (i.e., power uprates). For power uprates, 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 NRC's Review Standard (RS-001), "Review Standard for Extended Power Uprates," dated December 2003. Since the issuance of the last U.S. National Report, the NRC has reviewed and approved extended power uprates for Hope Creek Generating Station, Unit 1; Peach Bottom Atomic Power Station, Units 2 and 3; Browns Ferry Nuclear Plant, Units 1, 2, and 3; and Columbia Generating Station. Section 14.1.3 of this report provides additional information on power uprates.

12.2.2 Organizational Issues

In accordance with NUREG-0800, Chapter 13, "Conduct of Operations," the staff reviews a license applicant's (e.g., for a construction permit, operating license, standard design certification, combined license, or license transfer) corporate-level management and technical support organization. The review includes the applicant's major contractors, including the nuclear steam supply system vendor and architect-engineer for the project. The NRC also reviews the applicant's operating organization and technical resources to support the nuclear power plant design, construction, testing, and operation. The review includes the structure, functions, and responsibilities of the onsite organization established to safely operate and maintain the facility. Section 11.2 of this report provides additional information about qualification and training of plant personnel.

The NRC also reviews license amendment requests and other licensing action requests that propose changes to the licensee's management, technical and operating organizations for operating plants and plants transitioning to decommissioning. Examples include approvals of the licensee's certified fuel handler training program, amendments to plant technical specifications associated with administrative controls and staff qualifications, and orders consenting to transfer an operating license. Since the issuance of the last U.S. National Report, the NRC reviewed the approved certified fuel handler training programs for the Oyster Creek Nuclear Generating Station, Quad Cities Nuclear Power Station, and Palisades Nuclear Plant.

12.2.3 Emergency Operating Procedures and Plant Procedures

In accordance with Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, licensees develop, implement, and maintain emergency operating and plant procedures. Procedures guide the operators on how to respond in a way that provides for safe operation of the plant and are an important element in human factors considerations. NUREG-0800, Chapter 13, is used to review an applicant's plan for development and implementation of the operating procedures to ensure that routine operating, off-normal, and emergency activities are conducted safely. 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," requiring each licensee to submit a set of documents for developing emergency operating procedures.

The events at Fukushima Dai-ichi in March 2011 highlighted the need for power reactor licensees to have strategies for responding to beyond-design-basis external events affecting one or more units at a site. On March 12, 2012, the NRC issued Order EA-12-049 requiring licensees to develop these mitigation strategies. The nuclear industry proposed regulatory guidance, endorsed by the NRC, which outlines an approach for developing these strategies. The approach is called the Diverse and Flexible Mitigation Strategies (commonly known as “FLEX”) and is focused on maintaining or restoring key plant safety functions. This regulatory guidance provides a procedural approach for the implementation of FLEX strategies, which includes evaluating these strategies for integration with the appropriate existing procedures. This regulatory guidance also provides a method to validate the strategies to show they are feasible and that the personnel who would need to use the strategies in an actual event can execute them. In addition, the NRC requested that licensees assess their emergency communications systems and staffing levels to ensure that sufficient resources are available to respond to an event involving all units at each site.

A final rule, 10 CFR 50.155. “Mitigation of Beyond-Design-Basis Events” (84 FR 39684), was approved by the Commission, making the requirements of Order EA-12-049 generically applicable in the NRC’s regulations. As with Order EA-12-049, this rule requires licensees to develop, implement, and maintain strategies and guidelines to mitigate beyond-design-basis external events. In conjunction with this rule, in June 2019, the staff issued RG 1.226, “Flexible Mitigation Strategies for Beyond Design Basis Events.” This RG endorses, with certain exceptions and clarifications, the industry guidance for mitigation strategies, including the method for validation of time sensitive manual actions documented in Appendix E to NEI 12-06, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Revision 4, dated December 2016. Section 2.3.3.4 of this report provides further details of the requirements implemented as a result of the Fukushima lessons learned.

12.2.4 Shift Staffing

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.

Current staffing requirements are based on assumptions and operating experience from the operation of large light-water reactors. Also, the staffing requirements in 10 CFR 50.54(m) do not address a situation where three or more units are controlled from a single control room, which has been proposed by some designers of small modular reactors. Therefore, 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 guidance addresses the changing demands and new technologies presented by advanced reactor control room designs and significant light-water reactor control room upgrades.

A key element is the review of the applicant’s staffing plan validation, which is an evaluation using performance-based tests to determine whether the staffing plan meets performance requirements and acceptably supports safe operation.

The NRC staff is currently reviewing NuScale’s small modular reactor design certification application. The design includes control of up to 12 units from a single main control room. The applicant submitted the results of its staffing plan validation with the design certification application. The NRC is using the guidance in NUREG-1791 to review the results. The NRC has conducted pre-application activities related to shift staffing with a number of other small modular reactor designers. Recent experience indicates that some applicants may be challenged to establish simulation capabilities to support such validation activities while they are finalizing other aspects of the plant design.

12.2.5 Human Performance in the Reactor Oversight Process

The Reactor Oversight Process focuses on safety cornerstones that are assessed through a combination of performance indicators and risk-informed inspections. Section 6.3.2 of this report provides a discussion of the Reactor Oversight Process and its seven safety cornerstones. In addition to the safety cornerstones, the Reactor Oversight Process assesses licensee performance in three elements are addressed across the cornerstones: human performance, safety-conscious work environment, and problem identification and resolution.

Human factors experts participate in Reactor Oversight Process special inspections, incident investigation team inspections, augmented team inspections, event investigations, and supplemental inspections, as needed. Human factors experts assess management effectiveness, procedures, training issues, staffing issues, human-machine interfaces, personnel performance issues, safety-conscious work environment, and safety culture. Section 10.3 of this report provides more information about safety culture.

To evaluate licensees' procedures, NRC inspectors use IP 42001, "Emergency Operating Procedures," dated June 28, 1991, and IP 42700, "Plant Procedures," dated November 15, 1995. Weaknesses in problem identification and resolution programs may manifest themselves as performance issues that cross predetermined indicator thresholds. To address these types of issues, inspectors use IP 71152, which includes a review of the licensee's safety-conscious work environment to confirm that the licensee gives priority to maintaining safety.

NRC inspectors use IP 95003 to provide supplemental inspection response for plants with repetitive or multiple degraded cornerstones in the Reactor Oversight Process Action Matrix. The NRC revised IP 95003 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. NRC 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.

In 2017, human factors experts participated in the IP 95003 inspection at Pilgrim Nuclear Power Station using an updated version of IP 95003. The overall result and conclusion of the inspection was that the plant was being operated safely; however, the inspection found that the licensee needed to continue aggressive implementation of its action plan to achieve substantial performance improvement. The NRC has completed the required followup inspections and determined that plant performance was sufficient to move the plant from Column 4 to Column 1 of the Reactor Oversight Process Action Matrix.

Inspection findings associated with human performance or safety culture issues are used as inputs to an analysis tool called the Human Factors Information System, described in Section 12.2.6 of this report.

12.2.6 Human Factors Information System

The Human Factors Information System is designed to store, retrieve, sort, and analyze human performance information extracted from NRC inspections and LERs. Initiated in 1990, this 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. In 2016 and 2017, the NRC automated the inputs from various data sources to quickly compile reports to support various analyses. Currently, the agency is updating the database to include data with a safety culture perspective.

The NRC regularly responds to stakeholder and public inquiries and data requests on this system. For example, NRC inspectors use the data in the Human Factors Information System while 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 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: https://www.nrc.gov/reading-rm/doc-collections/human-factors/.

12.2.7 Fitness for Duty

In 10 CFR Part 26, "Fitness for Duty Programs," the NRC requires each power reactor licensee to 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 requires licensees and permit holders authorized to construct a nuclear power plant to implement a fitness for duty program for 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 are trustworthy and reliable as demonstrated by avoiding substance abuse

(2) Provide reasonable assurance that personnel are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause that in any way adversely affects their ability to safely and competently perform their duties

(3) Provide reasonable measures for the early detection of persons who are not fit to perform activities covered by 10 CFR Part 26

(4) Provide reasonable assurance that the workplaces are free from the presence and effects of illegal drugs and alcohol

(5) Provide reasonable assurance that the effects of fatigue on an individual's ability to safely and competently perform his or her duties are managed commensurate with maintaining public health and safety

In 2008, the NRC amended 10 CFR Part 26 to include specific provisions for the management of worker fatigue. RG 5.73, “Fatigue Management for Nuclear Power Plant Personnel,” dated March 2009, presents guidance for implementing Part 26, Subpart I, “Managing Fatigue.”

A major impetus for amending 10 CFR Part 26 to include fatigue management requirements was the extensive use of waivers to deviate from technical specifications limits on individual work hours. As noted in SECY-01-0113, “Fatigue of Workers at Nuclear Power Plants,” dated June 22, 2001, the number of deviations in 1999 during non-outage periods ranged from 12 to 992 per site, for the 40 sites that provided data. During outage periods in 1999, the range of authorized deviations was 7 to 7,553 per site. About one quarter of the sites authorized reported more than 2,000 deviations during outage periods. Following the amendment of 10 CFR Part 26 to include enforceable work hour limits, these numbers drastically reduced. In 2010, the first full year of implementation of the fatigue management requirements, the number of waivers authorized (including both operating and outage periods) averaged 38 per site for the 57 U.S. nuclear power plant sites reporting. In 2015, after several years of experience with the fatigue management requirements, this average dropped to approximately 5.5 for the 61 sites reporting. For 2017, 211 waivers were reported for 54 sites. For 2018, 253 waivers were reported for 58 sites (an average of approximately 4 waivers per site for the year).

12.3 Licensee Human Factors Programs

The NRC does not require licensees to maintain a specific program for human factors engineering, and therefore, the agency does not conduct associated programmatic inspections. Rather, in keeping with a risk-informed, performance-based approach to licensee oversight, the NRC evaluates the human factor engineering aspects of modifications to nuclear power plants, control rooms, and modifications affecting important human actions that are submitted to the NRC under 10 CFR 50.59. Similarly, the NRC does not require licensees to maintain specific programs for analyzing, preventing, detecting and correcting human errors in operation and maintenance. However, licensees implement programs that fulfill these functions consistent with the NRC’s quality assurance requirements. Specifically, 10 CFR Part 50, Appendix B, includes requirements for licensee managerial and administrative controls to be used to ensure safe operation. For example, the identification and correction of human errors in operation and maintenance are more broadly addressed under 10 CFR Part 50, Appendix B, Criterion XVI.

12.4 Feedback and Experience

As new technologies are introduced and regulatory issues emerge, the NRC updates its requirements and regulatory guidance documents to reflect feedback and experience. The following examples describe recent or current initiatives that address human performance considerations at nuclear facilities. Article 18 of this report discusses human factors in new plant design certifications.

12.4.1 Human Factors Associated with Digital Instrumentation and Control

To make the NRC’s human factors guidance simpler, clearer, and more relevant to the digital environment, the staff issued 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. In 2016, the staff updated the guidance for crediting of operator actions and incorporated this guidance as Appendix A, "Crediting Manual Operator Actions In Diversity and Defense-In-Depth Analyses," to Chapter 18 of NUREG-0800. To update its guidance on computer-based procedures, the NRC plans to develop an RG that, if approved, 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.

12.4.2 Human Performance in Decommissioning Activities

As discussed in Section 2.3.3.3 of this report, on May 7, 2018, the staff issued a paper (SECY-18-0055) requesting Commission approval of a draft proposed rule on decommissioning activities. In this paper, the NRC identified the certified fuel handler position, staffing levels, and training as potential areas for change. The certified fuel handler at a decommissioning reactor is the individual with the requisite knowledge and experience to evaluate plant conditions and make judgments about the actions necessary to protect public health and safety. The draft proposed rule would provide an alternative definition for the certified fuel handler. In addition, the draft proposed companion guidance, DG-1347 (proposed Revision 2 to RG 1.184, "Decommissioning of Nuclear Power Reactors"), includes specific criteria for certified fuel handler training programs to ensure the safe conduct of decommissioning activities, safe handling and storage of spent fuel, appropriate response to plant emergencies, and command and control over these functions.

12.4.3 Human Performance Research

The Human Performance Research Program generates, collects, and evaluates 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.

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.

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

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

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

Each applicant for a combined license under 10 CFR Part 52 must describe its quality assurance program in a safety analysis report and explain the managerial and administrative controls that will be applied 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

Under 10 CFR 50.34 and various provisions in 10 CFR Part 52, an application must include principal design criteria for a proposed facility. Appendix A to 10 CFR Part 50 provides general design criteria that establish the minimum requirements for principal design criteria for water-cooled nuclear power plants similar to previously licensed nuclear power plants. This includes details for the general requirements for establishing quality assurance controls. General Design Criterion 1, "Quality Standards and Records," addresses the quality assurance of items important to safety. The scope of items "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, Appendix B specifies 18 quality criteria that must be addressed in a licensee's quality assurance program description. These criteria cover such topics as organizational independence, design control, procurement, procedures, document control, test control, special processes, calibration, corrective action, quality assurance records, 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 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 developed 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, "Quality management systems – Requirements" by considering how international standards compare with the existing framework in Appendix B to 10 CFR Part 50. Based on 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, operating licenses, early site permits, or design certifications, and licensees. This guidance applies 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 in Appendix B to 10 CFR Part 50 and integrates a review of licensee commitments to adopt the NRC's quality assurance-related RGs and apply the industry's quality assurance codes and standards.

13.3.2 Guidance for Design and Construction Activities

Licensees may apply consensus standards developed by the American National Standards Institute (ANSI) in its N45.2 series or by ASME in its Nuclear Quality Assurance (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 5, dated October 2017, to endorse NQA-1b-2011 Addenda to ASME NQA-1-2008, NQA-1-2012, and NQA-1-2015.

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," dated February 1976, through RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, dated February 1978, as complying with the requirements of Appendix B to 10 CFR Part 50. Subsequently, the NRC staff issued RG 1.33, Revision 3, in June 2013, endorsing a newer standard, ANSI/ANS 3.2-2012, "Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants," dated March 20, 2012. ANSI/ANS 3.2-2012 focuses on quality assurance of plant operations because another standard contains information on quality assurance of design and construction.

13.4 Quality Assurance Programs

The NRC inspects quality assurance programs under the Reactor Oversight Process for operating reactors and under the Construction Reactor Oversight Process (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. NRC inspectors use this procedure to assess the effectiveness of licensees' programs to find and resolve problems through a performance-based review of specific issues. NRC 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 but 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 of SSCs. Typically, applying quality assurance controls to this important-to-safety, yet nonsafety-related, equipment is called "augmented quality control."

The Construction Reactor Oversight Process provides oversight for new nuclear plants permitted or 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 NRC inspectors use IP 35007, "Quality Assurance Program Implementation during Construction and Pre-Construction Activities," dated December 12, 2016, to verify that the holder of a construction permit or combined license has developed quality assurance procedures, instructions, and other documents that are consistent with the licensee's NRC-approved quality assurance program description and to verify that the permit holder or licensee has effectively implemented its quality assurance program implementing documents during construction activities.

Oversight of a new nuclear plant will transition from the Construction Reactor Oversight Process 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 safety-related 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, which 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 in accordance with 10 CFR Part 21, "Reporting of Defects and Noncompliance." Inspection Manual Chapter 2507, "Vendor Inspections," dated May 16, 2018, 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 accordance 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 addresses the Vienna Declaration on Nuclear Safety, issued February 2015.

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 many 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.

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. A licensee must continue to meet its current licensing basis during the period of extended operation following license renewal; 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: (1) reactor safety, (2) radiation safety, and (3) safeguards. 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 manner like that used by the Reactor Oversight Process. The NRC monitors plant construction in three key areas: (1) construction reactor safety, (2) operational readiness, and (3) safeguards programs. Inspection findings are used to assess construction 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's regulatory programs are in place to provide 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 a licensing basis, as required by 10 CFR 50.54, 10 CFR 50.59; 10 CFR 50.71, "Maintenance of Records, Making of Reports"; and 10 CFR Section 50.90, "Application for Amendment of License, or Construction Permit, or Early Site Permit."

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, as updated. To change its license or reactor facility, a licensee must follow the review and approval processes established in the regulations. For changes to the operating license or combined license, including changes to technical specifications, the licensee must submit an amendment request for NRC approval in accordance with 10 CFR 50.90. However, 10 CFR 50.54 and 10 CFR 50.59 (see below) contain requirements for the process by which, under certain conditions, licensees may make changes to their facilities and procedures as described in the final safety analysis report, as updated, without prior NRC approval. Additionally, 10 CFR 50.54 contains requirements for the processes by which, under certain conditions, licensees may make changes to specific portions of their licensing basis without prior NRC approval. For combined license holders that reference a certified design, a comparable process for changes and departures from information within the scope of the referenced design certification rule is described in the applicable Appendices to 10 CFR Part 52.

10 CFR 50.54(a). In 10 CFR 50.54(a), the NRC establishes the conditions under which a licensee may make changes to its previously accepted quality assurance program description without prior NRC approval if the changes do not reduce the commitments in the program description accepted by the NRC and the changes are submitted to the NRC in accordance with 10 CFR 50.71 for periodic final safety analysis report updates.

10 CFR 50.54(q). In 10 CFR 50.54(q), the NRC establishes the conditions under which a licensee may make changes to its emergency plan without prior NRC approval if the licensee performs and retains an analysis demonstrating that the changes do not reduce the effectiveness of the plan, and if the plan, as changed, continues to meet the requirements in Appendix E to 10 CFR Part 50 and, for nuclear power reactor licensees, the planning standards of 10 CFR 50.47(b).

RG 1.219, Revision 1, "Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors," dated July 2016, describes a method the NRC staff considers acceptable for implementing the requirements of 10 CFR 50.54(q).

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 it 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,” Revision 1, dated May 2019, endorsed, with clarifications, industry guidance document NEI 96-07, Revision 1, “Guidelines for 10 CFR 50.59 Evaluations,” dated February 2000, which provides a method that is acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59.

10 CFR 50.71. In 10 CFR 50.71, the NRC establishes requirements for licensees to update their final safety analysis reports periodically to incorporate the information and analyses that they submitted to the Commission. Revisions to the 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 or 10 CFR 52.98, “Finality of Combined Licenses: Information Requests,” as applicable, and safety analyses conducted at the request of the Commission to address new safety issues.

RG 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e),” dated September 1999, endorsed NEI 98-03, “Guidelines for Updating Final Safety Analysis Reports,” Revision 1, dated June 1999, as an acceptable method for complying with the provisions of 10 CFR 50.71(e).

10 CFR 50.90. Under 10 CFR 50.90, whenever a holder of a license, including a construction permit and operating license under 10 CFR Part 50, or an early site permit, combined license, or manufacturing license under 10 CFR Part 52, wants to amend the license or permit, it must file an application for an amendment with the Commission. The NRC specifies the requirements for filing in 10 CFR 50.4 or 10 CFR 52.3, both titled “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, and issues an amendment in these instances before it authorizes the change.

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. The license and technical specifications for the plant include this power level. 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, called a power uprate.

Categories of Power Uprates. The NRC has specified three categories of power uprates:

(1) Measurement Uncertainty Recapture Power Uprates—These 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, which 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.

(2) Stretch Power Uprates—These 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.

(3) Extended Power Uprates—These 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, a licensee must seek NRC approval to amend its 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 whether the plant can operate safely at the higher power level.

The NRC uses RS-001 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," dated January 31, 2002, discusses the scope and detail of the information that should be provided to the NRC for reviewing measurement uncertainty recapture uprate applications. Additionally, the staff uses NUREG-0800, where appropriate, when conducting power uprate regulatory reviews.

After a licensee submits an uprate application, the NRC issues a *Federal Register* notice to alert the public that the agency is considering the application. The public has 30 days to comment on the licensee's request and 60 days to request a hearing where the application could be contested. The NRC thoroughly reviews the application and any public comments, while the Atomic Safety and Licensing Board considers any requests for hearings. The NRC documents its review in a safety evaluation, and, if acceptable, the NRC will issue a license amendment approving the power uprate. The NRC will issue another *Federal Register* notice to inform the public if the amendment is issued. After the approval, the NRC inspects the power uprate implementation using IP 71004, "Power Uprate," dated May 15, 2017, to review plant modifications and operator readiness.

If the Atomic Safety and Licensing Board determines that a hearing is required, a separate legal proceeding takes place, and the NRC staff provides technical information, if needed. The safety evaluation and any final rulings from the adjudicatory hearing process form the basis for the NRC's final decision on the uprate request. However, the staff can authorize an uprate before the adjudicatory proceedings are completed but may need to modify or further amend the license to reflect the results of the hearing. The NRC issues a press release for any approved uprate.

The NRC's expected schedule is to complete power uprate reviews within 18 months of accepting the application for review for extended power uprates, within 12 months of acceptance for stretch power uprates, and within 9 months of acceptance for measurement uncertainty recapture uprates. The application acceptance process is intended to give the NRC staff an opportunity to ensure that application quality is sufficient for the detailed safety review to begin.

14.1.3.2 Experience

The NRC issued the first power uprate amendment for the Calvert Cliffs Nuclear Power Plant in 1977. As of August 2019, the NRC had approved 164 uprates, resulting in a gain of approximately 23,800 megawatts thermal (MWt) or 7,900 megawatts electric (MWe), at existing plants.

Since the issuance of the previous U.S. National Report, the NRC has approved four measurement uncertainty recapture power uprates totaling 250 MWt (83 MWe) of generational capacity for Columbia Generating Station; Peach Bottom Atomic Power Station, Units 2 and 3; and Hope Creek Generating Station, Unit 1.

The NRC also issued three extended power uprates for Browns Ferry Nuclear Plant, Units 1, 2 and 3. The extended power uprate for each unit remained within 120 percent of the original licensed thermal power level. Multiple major plant modifications and replacements were associated with the power uprate, including improvements in the design of the steam dryers.

The NRC approval of the extended power uprate for Browns Ferry was based, in part, on the licensee's capability to monitor, evaluate, and act promptly 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. As part of the extended power uprate amendment, the NRC added numerous license conditions to the facility operating license for each unit, to provide the necessary requirements associated with potential adverse flow effects and power ascension to the newly approved licensed thermal power level.

For Browns Ferry Unit 3, the licensee completed the plant modifications needed to implement the extended power uprate during the fall 2017 refueling outage, including installation of the replacement steam dryer. During the power ascension following the refueling outage, data collected at various hold points up to the newly approved power level identified acceptable strain responses on the replacement steam dryer at all measured frequency ranges. For Unit 3, the licensee reached the new 100 percent extended power uprate power level on July 13, 2018.

Browns Ferry, Unit 1, achieved the extended power uprate full power in February 2019. Browns Ferry, Unit 2, achieved the extended power uprate full power in June 2019.

The NRC currently anticipates that U.S. licensees will submit five measurement uncertainty recapture power uprate applications within the next 2 years. If these expected applications are approved, the resulting uprates would authorize an additional 293 MWt (98 MWe). Additional information about power uprates can be found on the NRC's public Web site at https://www.nrc.gov/reactors/operating/licensing/power-uprates/status-power-apps/approved-applications.html.

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 based on economic and antitrust considerations rather than technical limitations; however, many of the technical safety evaluations were subsequently based on a 40-year operating period. 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 established a license renewal process with requirements to ensure safe plant operation for up to 20 additional years. The NRC's expected schedule is to complete the review of a license renewal application within 18 months of acceptance of the application if an adjudicatory hearing is not conducted. If there is a hearing, the agency will adjust the schedule in accordance with 10 CFR Part 2, as necessary.

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 related to aging 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 proposed period of extended operation. 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 should any party that would be adversely affected seek a hearing.

10 CFR Part 54. Known as the License Renewal Rule, 10 CFR Part 54 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 each currently operating plant provides an acceptable level of safety.

(2) Each plant must maintain its current licensing basis throughout the renewal term, with the addition of aging management activities.

These principles ensure that safety continues to be maintained during the license renewal period of extended operation. The 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, dated September 2005, which guides applicants preparing an application for a renewed license, and NUREG-1800, Revision 2, dated December 2010, which guides the staff in reviewing applications. The standard review plan for license renewal incorporates by reference NUREG-1801, Revision 2, dated 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 or in license renewal ISG documents.

NUREG-1801 is a technical basis document, which provides the staff with guidance in reviewing a license renewal application. It provides generic evaluations of the aging effects that require aging management, and describes acceptable aging management programs (considering the materials and environment for each SSC). An applicant may reference NUREG-1801 in a license renewal application to demonstrate that the programs at the applicant's facility correspond to those reviewed and endorsed by the NRC.

If an applicant takes credit for a program in NUREG-1801, the applicant must ensure that the plant's aging management program contains the following 10 elements:

(1) Scope of the Program—The scope of the program should include the specific structures and components subject to an aging management review.

(2) Preventive Actions—Preventive actions should mitigate or prevent the applicable aging effects.

(3) Parameters Monitored or Inspected—Parameters monitored or inspected should be linked to the effects of aging on the intended functions of the structure and component.

(4) Detection of Aging Effects—Aging effects should be detected before there is a loss of any structure and component intended function. This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of new or one-time inspections to ensure timely detection of aging effects.

(5) Monitoring and Trending—Monitoring and trending should provide for prediction of the extent of the effects of aging and timely corrective or mitigative actions.

(6) Acceptance Criteria—Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure and component's intended functions

are maintained under all current licensing basis design conditions during the period of extended operation.

(7) Corrective Actions—Corrective actions, including root cause determination and prevention of recurrence, should be timely.

(8) Confirmation Process—The confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

(9) Administrative Controls—Administrative controls should provide a formal review and approval process.

(10) Operating Experience—Operating experience involving the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the structure and component intended functions will be maintained during the period of extended operation.

NUREG-1801 contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for the NRC staff to review in its plant-specific license renewal application. The use of NUREG-1801 is not required, but its use should facilitate both preparation of the license renewal application by an applicant and timely, uniform, and complete review by the NRC staff.

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 environmental review requirements for license renewal 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 10 CFR 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 the NUREG was issued in 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 to 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 more 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 Plant 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, Revision 1, "Standard Review Plans for Environmental Reviews of Nuclear Power Plants: Environmental

Standard Review Plan for Operating License Renewal," guides the NRC staff's review of the environmental issues associated with license renewal.

The 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 agency 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 operating period in the environmental analysis.

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 January 1, 2019, 93 reactors have received renewed licenses.[10] By the end of 2018, 49 of the 89 operating reactors with renewed licenses that are still currently operating have completed 40 years of operation and are operating in the extended period. Six reactor units entered the period of extended operation between 2017 and 2018 and three reactor units are expected to enter the period of extended operation in 2020. Based on industry statements, the NRC expects that all but two of the remaining units that have yet to tender license renewal applications will apply for license renewal. For a list of plants that are expected to apply for license renewal, see
http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html.

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 without additional requirements and with 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 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).

To prepare for the review of subsequent license renewal applications, on January 31, 2014, the NRC staff submitted SECY-14-0016, "Ongoing Staff Activities To Assess Regulatory Considerations for Power Reactor Subsequent License Renewal," to the Commission. 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 do the following:

- Continue to update the license renewal guidance and address emerging technical issues and operating experience.

10 After receiving their renewed licenses, Kewaunee Power Station, Vermont Yankee Nuclear Power Station, Fort Calhoun Station, Oyster Creek Nuclear Generating Station, and Pilgrim Nuclear Power Station, Unit 1, permanently ceased operations. As of May 2019, these plants had announced that they would cease operations before the expiration of their renewed license: Three Mile Island Nuclear Station; Davis-Besse Nuclear Power Station; Beaver Valley Power Station; Indian Point Nuclear Generating; Duane Arnold Energy Center; and Palisades Nuclear Plant.

- Keep the Commission informed of the staff's progress in resolving technical issues and the staff's readiness to accept an application and any further need for regulatory process changes, rulemaking, or research related to subsequent license renewal.

On July 10, 2017, the staff met the Commission's direction by publishing the NUREG-2191 and NUREG-2192. These final guidance documents support the staff's readiness to receive and evaluate the acceptability of a subsequent license renewal application. NUREG-2191 provides guidance on the content of subsequent license renewal applications and identifies acceptable methods to manage aging effects for nuclear plant operations from 60 to 80 years of operation. NUREG-2192 provides guidance to the staff performing safety reviews of these applications.

As requested, the staff periodically met with the Commission and informed them about progress in resolving technical issues. For example, the staff cooperated with DOE and EPRI to address the four key technical issues outlined in SRM-SECY-14-0016: (1) reactor pressure vessel neutron embrittlement at high fluence, (2) irradiation assisted stress corrosion cracking of reactor internals and primary system components, (3) concrete and containment degradation, and (4) electrical cable qualification and condition assessment. Consistent with Commission direction "to strive for satisfactory resolution of these issues prior to the NRC beginning a review of any SLR application," the staff worked extensively with industry and other external stakeholders to incorporate appropriate guidance on these issues in NUREG-2191 and NUREG-2192. As discussed in Section 2.3.1.9 of this report, the NRC continues to perform long-term confirmatory research that will provide additional generic information that will make reviews more effective and efficient as additional licensees submit subsequent license renewal applications. Sections 2.3.1.9 and 2.3.2.9 of this report describe the NRC's experience with subsequent license renewal.

14.1.5 The United States and Periodic Safety Reviews

Many countries conduct periodic safety reviews (typically carried out every 10 years) consistent with the 2013 IAEA Specific Safety Guide SSG-25, "Periodic Safety Review for Nuclear Power Plants," to assess safety factors, including 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 and subsequent license renewal processes, are considered equally adequate and acceptable.

This section explains how the U.S. regulatory approach provides continuous assessment and review that ensures public health and safety throughout the period of plant operation. Plant safety is maintained, and aspects are improved, during its initial licensing period, license renewal, and subsequent license renewal, through a combination of the ongoing NRC regulatory process, oversight of the current licensing basis, backfitting, broad-based evaluations, 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 requirements and provide reasonable assurance of 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, taken together, constitute a process offering 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 continues to provide reasonable 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.

The scope of license renewal includes (1) safety-related SSCs, (2) all nonsafety-related SSCs whose failure could adversely impact safety functions, and (3) all SSCs relied on in certain safety analyses or plant evaluations for specific NRC regulations. The focus of the license renewal review is on aging management of long-lived, passive structures and components in nuclear power plants (e.g., reactor pressure vessel, steam generators, piping, seismic Category I structures, and electrical cables and connections). The structures and components that are not subject to aging management are those that have active functions, such that their failure would be identified during surveillance and testing in accordance with 10 CFR 50.65 or replaced on a fixed schedule. The regulation in 10 CFR 50.65 focuses on monitoring and testing activities to ensure that SSCs can perform their intended functions.

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 NRC regulations such as those described in 10 CFR 50.54, 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 made to the licensing basis that did not require NRC review and approval at least every 2 years. As stated in 10 CFR 50.71(e), the periodic update ensures that the final safety analysis report contains the latest information on the facility's licensing basis. 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. Because these activities are critical to the agency's mission, the NRC devotes considerable resources to the oversight process. For example, each plant receives 6,000 to 10,000 hours of inspection every year. Additionally, the NRC staff spends more than 1,200 hours evaluating licensing tasks at each plant. This level of effort gives the Commission the confidence that the oversight process ensures public health and safety and 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 of the Reactor Oversight Process.

14.1.5.2 The Backfitting Process: Timely Imposition of New Requirements

In the 1960s, as nuclear energy technology was rapidly developing, the NRC recognized the need for a process to determine when to require licensees to install improved safety features in facilities that were under construction or operating. As a result, the NRC developed the "backfitting" process and, in 1981, established the Committee to Review Generic Requirements to review proposed backfits on licensees.

The Backfitting Rule, 10 CFR 50.109, first issued in 1970 and substantially revised in 1985 and 1988, applies to both generic and plant-specific backfitting for power reactors. The rule applies to any modification of or addition to (1) facility systems, (2) facility structures, (3) facility components, (4) facility designs, (5) design approvals, (6) manufacturing licenses, or (7) procedures or organization required to design, construct, or operate a facility—any of which may result from the imposition of a new or amended rule or regulatory staff position. In 1989, the NRC extended backfitting-style provisions to nuclear power plants licensed under 10 CFR Part 52. These 10 CFR Part 52 procedures, referred to as issue finality, function similarly to backfitting requirements and provide a rigorous process for determining when the NRC can impose new requirements on previous approvals, including early site permits, standard design certifications, and combined licenses. The NRC also put in place backfitting provisions for independent spent fuel storage installations, gaseous diffusion plants, and major fuel cycle facilities in 1988, 1994, and 2000, respectively.

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 backfitting process includes a Committee to Review Generic Requirements review, which is a committee of senior managers from different NRC offices. This committee operates under a charter that specifically identifies the documents that will be reviewed. Its primary responsibilities are to (1) recommend to the NRC's Executive Director for Operations either approval or disapproval of staff proposals related to backfitting and (2) provide guidance and assistance to the NRC program offices to help them implement the Commission's backfitting policy. Therefore, the review by the Committee to Review Generic Requirements is a key step in implementing the NRC's backfitting process, although the primary responsibility for proper backfitting considerations belongs to the NRC staff initiating the backfitting action.

14.1.5.3 License Renewal Confirms Safety of Plants

In developing the License Renewal Rule (10 CFR Part 54) in 1995, the Commission concluded that issues material to the renewal of a nuclear power plant operating license are limited to those issues that 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 and are dealt with during the current plant operating period. Given the Commission's ongoing obligation to oversee the safety and security of operating reactors, the existing regulatory process under a licensee's current license 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 license renewal process focuses on aging management of passive and long-lived SSCs because degradation in active components is more readily detected by complying with the Maintenance Rule (10 CFR 50.65) as discussed in Section 14.1.5.1 of this report. License renewal applicants are required to complete an environmental assessment and an integrated plant assessment[11] 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.

14.1.5.4 Risk-Informed Regulation and the Reactor Oversight Process

The NRC has incorporated the use of risk insights and risk information in its regulatory decisionmaking processes. A risk-informed approach to regulatory decisionmaking considers risk insights together with other factors to establish requirements and guide oversight, with the goal of focusing 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) 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 regulations, risk-informing technical specifications, updates to inspection and assessment processes, 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, regulations, and oversight approaches. Significant plant operating events occurred with some frequency, and the oversight process tended to be reactive and prescriptive, observing plant performance for adherence to the regulations and responding to operational problems as they occurred.

After more than five decades of operational experience, the Reactor Oversight Process now focuses the agency's resources on issues based on their safety significance and on the relatively few plants requiring additional regulatory attention based on their performance. In general, the Reactor Oversight Process provides for the collection of information about licensee performance, assessment of this information for its safety significance, and guidance for appropriate NRC response, including additional inspections and enforcement actions, when appropriate.

The Reactor Oversight Process uses direct NRC inspections and objective performance indicators reported by the licensee to measure and assess plant performance. Together, the

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11 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.

performance indicators and inspection findings give the information needed to support relevant and timely assessments of plant performance. The Reactor Oversight Process also features comprehensive quarterly reviews and expanded annual reviews, which include inspection planning and performance reporting (all posted on the NRC's public Web site). The Reactor Oversight Process is more effective at correcting plant performance and equipment problems today because the agency's response to problems is focused and predictable. Section 6.3.2 of this report fully describes the NRC Reactor Oversight Process.

14.1.5.5 Licensee Responsibilities for Safety: Regulations and Initiatives beyond Regulations

As in many countries, U.S. nuclear power plant licensees are ultimately responsible for the safety of their facilities. This responsibility is embedded in their license and in the NRC's regulatory framework. Under the regulatory umbrella, licensees routinely assess new technologies, off-normal conditions, operating experience, and industry trends to make informed decisions about safety enhancements to their facilities.

Under the U.S. regulatory structure, Appendix B to 10 CFR Part 50 requires 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. Appropriately trained personnel who do not have direct responsibilities in the areas being audited perform the audits in accordance with written procedures or checklists. Management reviews the audit results and initiates appropriate followup action.

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

The IAEA and the Western European Nuclear Regulators' Association have developed guidance[12] 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 following:

- 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 international safety standards and operating practices

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12 IAEA guidance appears in the 2013 Specific Safety Guide SSG-25. The Western European Nuclear Regulators' Association 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," Western European Nuclear Regulators' Association Reactor Harmonization Working Group, dated March 2013.

- safety improvements and timescales for their implementation
- the extent to which the safety documentation, including the licensing basis, remains valid

The 2010 IRRS mission in the United States concluded that the NRC processes sufficiently meet the objectives of periodic safety reviews and suggested that the NRC examine periodic safety review results from other countries. After the 2010 IRRS mission, 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 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 the NRC to stay apprised of international experiences in assessing reactor safety. The NRC welcomes such discussions during bilateral and multilateral exchanges as appropriate. Section 8.1.5.2 of this report provides additional information on the 2010 IRRS mission and the 2014 followup IRRS mission.

For the reasons discussed above and summarized below, the United States substantively accomplishes on an ongoing basis the shared objectives associated with periodic safety review guidance from IAEA and Western European Nuclear Regulators' Association.

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 (10 CFR 50.65) and License Renewal Rule (10 CFR Part 54) 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. Article 16 and Section 18.3.2.3 of this report discuss in more detail the NRC's actions in response to the attacks of September 11, 2001, in the areas of safety and security interface and cyber security.

Separately, the NRC has taken significant actions to enhance the safety of nuclear power reactors in the United States 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 BWRs with Mark I and II containments; reevaluation of seismic and flooding hazards using current guidance, methods, and information; and enhancing SFP instrumentation. Section 2.3.3.4 of this report gives additional details on the NRC Fukushima-related accomplishments.

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 systematic evaluation program, the integrated safety assessment program, 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 make safety improvements. For more than 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 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 licensees' self-described motivation to pursue excellence and from the recognition that safety and economics are directly linked in the competitive, free-market U.S. energy industry. Part 3 of this report discusses the role of INPO in supporting the U.S. commercial nuclear power industry's focus on nuclear safety.

14.2 <u>Verification by Analysis, Surveillance, Testing, and Inspection</u>

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 BP&V Code and the ASME Operation and Maintenance of Nuclear Power Plants Code.

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 to take corrective action.

Under special circumstances, to ensure the safe operation of plants, the Commission may require under 10 CFR 50.54(f) that licensees submit written statements to the Commission. The Commission can use the written statements to determine whether the license should be modified, suspended, or revoked. For example, the NRC invoked the 10 CFR 50.54(f) requirements following the Fukushima Dai-ichi accident by issuing letters to obtain written information on current seismic and flooding hazard protection, seismic and flooding hazard reevaluations using up-to-date methods, and emergency preparedness communications and staffing capabilities. This information was used to determine if additional regulatory actions were needed to ensure public health and safety. Section 2.3.3.4 of this report provides additional details on the implementation of lessons from the Fukushima Dai-ichi accident.

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. Section 6.3.11 of this report also discusses the generic communication tools that the NRC uses to share operating experience and information on regulatory and technical matters.

14.3 Vienna Declaration on Nuclear Safety

On February 18, 2015, the contracting parties to the CNS issued the Vienna Declaration on Nuclear Safety in INFCIRC 872. The declaration does not establish new requirements but recommits the contracting parties to the implementation of the CNS principles and objectives to prevent accidents and mitigate radiological consequences, as discussed in Articles 6, 14, 17, 18, and 19. Section 2.4.1.2 of this report summarizes the United States' implementation of these CNS objectives.

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 regarding radiation protection, the applicable regulatory framework for radiation protection, and certain measures for controlling radiation exposure to occupational workers and members of the public.

15.1 Overview of Regulatory Requirements and Authority

The United States has developed regulations for radiation protection to implement three key laws passed by the U.S. Congress: the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974; and the Uranium Mill Tailings Radiation Control Act of 1978. The U.S. approach to radiation protection is generally founded on radiological risk assessments conducted 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. These assessments reflect the risk management recommendations of the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements. Responsible agencies, such as the NRC, used these assessments, recommendations, and applicable laws, along with guidance from the executive branch, to establish regulations using a process that included and encouraged public participation. In summary, the primary authority of the NRC's regulations evolves from laws passed by Congress and is supported by the assessments of international and domestic scientific institutions.

NRC radiation protection regulations are based on principles comparable with those recommended by ICRP: limitation, justification, and optimization. Of these principles, "limitation" is the most evident in the NRC's regulatory structure. The regulations establish dose limits that if exceeded result in enforcement actions. "Justification" is the principle that any activity involving radiation exposure should be shown to be beneficial before the activity is undertaken. In the United States, the principle of "justification" is implemented during the licensing processes under 10 CFR Part 50 and 10 CFR Part 52 and during the operations phase through oversight.

Rather than using the term "optimization," the United States uses the term "ALARA" (the acronym for "as low as is reasonably achievable"). This use of ALARA (with varying terminology for this acronym) as a guiding principle dates to 1939. Before 1991, 10 CFR Part 20 addressed the ALARA criterion for occupational radiation exposure, but more as a recommendation than as a requirement. In 1991, the NRC revised 10 CFR Part 20 to require that all licensees use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. The NRC evaluates 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 represent an absolute minimum or whether the licensee used all possible methods to reduce exposures.

15.2 Regulatory Framework and Expectations

As Article 6 of this report discusses, 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 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's dose within the regulatory limits and occupational exposures to radiation that are ALARA. The Reactor Oversight Process evaluates licensee performance including compliance with regulations in a risk-informed, performance-based manner.

The regulations that apply to public and occupational radiation protection from nuclear power plant operations are 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." The NRC has additional requirements for specific operations and specific kinds of licenses in other parts of Title 10.

10 CFR Part 20. The NRC regulations in 10 CFR Part 20 establish requirements for radiation protection for all NRC licensees. The most recent 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," issued in 1978-1982, as well as some recommendations from the 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. Each subpart of 10 CFR Part 20 addresses a specific area of radiation protection, such as occupational and public dose limits, posting, surveys, monitoring, waste disposal, and reporting requirements.

Although U.S. regulations are generally consistent with ICRP recommendations, there are certain considerations that have limited the extent to which U.S. regulations match those of ICRP. One important factor has been the U.S. desire for regulatory stability as reflected in the Principle of Good Regulation concerning reliability. While the NRC staff regularly reviews new ICRP recommendations for applicability to existing guidance documents, NRC's position is that revising the regulations to incorporate every new ICRP recommendation would impose burdens on licensees without commensurate safety benefits. Licensees have the ability to request and use newer ICRP recommendations, following approval by the NRC, through license exemption requests. Another important consideration for U.S. nuclear power reactors, is that new requirements must be cost-justified and must provide a substantial increase in safety or must be needed to maintain adequate protection of public health and safety.

Similarly, 10 CFR Part 20 is generally consistent with international standards such as IAEA General Safety Requirements Part 3 (GSR-3), "Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards—General Safety Requirements," dated November 2014, with some notable differences: (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 sieverts (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 NRC approval. Many NRC licensees have administrative dose limits similar to, or lower than, those in the IAEA Basic Safety Standards. In fact, most licensees operate at occupational doses far below those standards. In rare cases, the occupational doses do exceed 0.02 Sv (2 rem) per year, but these are a very small fraction of the total, and licensees continue efforts to reduce doses as noted by the U.S. industry's collective dose performance over recent history. Section 15.3.1 of this report provides additional information on measured occupational exposure.

10 CFR Part 50. The regulations in 10 CFR 50.34(b)(3), 10 CFR 50.34(h), 10 CFR 50.34a, "Design Objectives for Equipment To Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors," and Appendix I, to 10 CFR Part 50, "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," require the NRC to review 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. 10 CFR 50.34(a)(1)(ii)(D) contains the revised dose criteria, in effective dose equivalent, for evaluating design basis accidents associated with licensing actions that have been submitted to the NRC since 1997. 10 CFR 100.11(a) contains the dose criteria used before 1997 for siting and determining the exclusion area low population zone and population center distance for nuclear power reactors.

10 CFR Part 71. The regulations in 10 CFR Part 71 apply to the transportation of licensed radioactive material. This regulation also sets procedures and standards for NRC review of packaging and transportation of radioactive material in excess of a Type A quantity. Finally, this regulation applies the U.S. Department of Transportation's rules for transportation of radiological material on NRC licensees; these regulations are found in 49 CFR Parts 107, 171 through 180, and 390 through 397, as applicable. The U.S. regulatory framework for the transportation of radioactive material is founded on the standards in IAEA's 2009 Specific Safety Requirements (SSR)-6, "Regulations for the Safe Transport of Radioactive Material. The NRC is actively working with the U.S. Department of Transportation to harmonize the current regulations with more recent versions of SSR-6.

15.3 Radiation Protection Activities and Control of Radiation Exposure

NRC radiation protection regulations recognize two fundamental characteristics of ionizing radiation: (1) doses of ionizing radiation above certain thresholds may result in nonstochastic health effects (e.g., cataract formation), and (2) there is an assumption about a direct and proportional relationship between radiation exposure and cancer risk with all radiation doses (known as the Linear No-Threshold Dose-Response Model). Radiation protection requirements apply to workers and members of the public and these requirements limit exposures from radiation to prescribed limits and achieving doses that are ALARA.

The NRC's oversight of radiation protection programs ensures that these programs satisfy all applicable requirements in a risk-informed and performance-based manner. The NRC maintains an active assessment process that consists of performance indicators and inspections.

Performance indicators provide quantitative measures of particular attributes of licensee performance that show how a plant is performing when measured against established thresholds. The inspection program includes routine baseline inspections and supplemental inspections, as needed. Additionally, any significant health physics issues that arise can result in reactive inspections.

The NRC documents histories of occupational exposures (NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 2016,” Volume 38, “Forty-Ninth Annual Report,” dated May 2018) and exposures to members of the public living near nuclear power plants (NUREG/CR-2907, “Radioactive Effluents from Nuclear Power Plants Annual Report 2014,” Volume 20, dated November 2018) as evidence that NRC requirements are adequate in this area and that licensee programs are sufficiently protective of workers and the public. More recent effluent release data are available on the NRC Web site at https://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

15.3.1 Control of Radiation Exposure of Occupational Workers

The NRC staff has been collecting the annual occupational exposure data for light-water reactors since 1969. Since the amount and type of maintenance performed strongly influence the doses, the individual plant collective doses fluctuate from year to year. As a result, in recent years the NRC has used a 3-year rolling average in communications about individual plant collective doses.

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 approximately a factor of 10, to the current level of about 0.74 person-Sv (74 person-rem) per reactor, based on a 3-year rolling average basis.

In 2017, 109,115 workers at nuclear plants were monitored for radiation exposure. In 2017, the median collective dose for BWRs and PWRs was 1.18 person-Sv (118 person-rem) and 0.37 person-Sv (37 person-rem), respectively. Of the monitored workers, 46,233 received a collective measurable dose of 64.17 person-Sv (6,417 person-rem) for an average of 0.00139 Sv (0.139 rem) per worker. Of the workers that received a measurable dose in 2017, 84 percent received less than 0.0025 Sv (0.25 rem), and 99.9 percent received less than 0.02 Sv (2 rem).

15.3.2 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,” limit 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 radiation sources associated with the entire uranium fuel cycle does not exceed 0.25 millisievert (25 millirem) to the whole body and 0.75 millisievert (75 millirem) to the thyroid. The regulations for license termination in 10 CFR Part 20, Subpart E, also state a 0.25 millisievert (25 millirem) limit, which is applicable to the average member of the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity (i.e., the critical group).

Additionally, regulations in 10 CFR 20.1406, 10 CFR 50.34a, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 contribute to controlling radiation exposure to members of the public by requiring licensees to minimize, to the extent practical, onsite residual radioactivity and radioactivity in effluents. Licensee programs to satisfy Appendix I to 10 CFR Part 50, 10 CFR 50.34a, and 10 CFR 50.36a provide data on the quantities of radioactive material released in effluents and material in the environment to evaluate the relationship between radioactive material released in effluents and the resultant doses to individuals from principal pathways of exposure. Additionally, licensees identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs. Appendix I requirements for ALARA are complemented by 10 CFR 20.1501, which requires, in part, that a licensee perform surveys, including those of the subsurface, to evaluate potential radiological hazards and to demonstrate compliance with public dose limits.

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 https://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.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.

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 emergency planning in the United States, including national response considerations, offsite emergency planning and preparedness, emergency classification system, inspection practices, and communications activities.

16.1 <u>Emergency Plans and Programs</u>

16.1.1 Background and Overview of Regulatory Requirements

The NRC's responsibilities for radiological emergency preparedness stem from the agency's licensing functions under the Atomic Energy Act and the Energy Reorganization Act. Both statutes authorize the Commission to issue regulations that it deems necessary to fulfill its responsibilities under the acts. After the accident at Three Mile Island Nuclear Station, 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 in the NRC's defense-in-depth safety philosophy of multiple-barrier containment and redundant safety systems. Before a full-power operating license can be issued, NRC regulations require a finding that there is reasonable assurance that adequate measures to protect public health and safety can and will be taken in a radiological emergency (10 CFR 50.47(a)).

Emergency planning in the United States recognizes that a spectrum of accidents could exceed the design-basis accidents that nuclear plants are required to accommodate without significant public health and safety effects. For design-basis accidents, the small releases that might occur would not likely require responses such as evacuating or sheltering the general public. These actions become important only when considering accidents that are much less probable than design-basis accidents. NUREG-0396, “Planning Basis for the Development of State and Local

Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," dated 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, dated November 1980, describe the emergency planning basis. These criteria provide a basis for licensees and states, tribal, and local governments to develop radiological emergency plans. NUREG-0654/FEMA-REP-1 is being revised to align with the NRC emergency preparedness rule changes and with the FEMA Radiological Emergency Preparedness Program Manual. As of July 2019, this joint document is with the Office of Management and Budget for clearance. The NRC expects to issue it after clearance is granted. The NRC revised its emergency preparedness regulations to address lessons learned from the September 11, 2001, terrorist attacks and security events. Section 16.1.2 of this report discusses this further.

After the Fukushima accident in March 2011, the NRC acted to further enhance emergency preparedness for licensees with respect to communications and staffing for responding to beyond-design-basis external events. In March 2012, the NRC asked licensees to evaluate their current communications systems and equipment, including appropriate enhancements, that would be used during an emergency event assuming that a large-scale natural event resulted in a loss of all AC power (i.e., a prolonged station blackout) and that cellular and other communications infrastructures were unavailable. Licensees also were asked to evaluate their emergency response organization staffing following the occurrence of a large scale natural event that altered the normal access routes to the site, thereby affecting the response time for the emergency response organization. As part of the assessment of their emergency response organizations, the licensees were asked to evaluate their current staffing levels and the appropriate staff and positions needed to respond to a multiunit event given a beyond-design-basis natural event and to determine if changes were needed. All licensees submitted the requested communications and staffing assessments, and the NRC responded to document the staff's reviews by July 2013 and March 2017, respectively.

16.1.2 National Response to an Emergency

The response to national emergencies fundamentally changed as a result of the publication of the National Response Framework in May 2013 and the update of its associated annexes. Additionally, DHS revised and republished the National Incident Management System (NIMS) document in October 2017. NIMS, which applies to all incidents, regardless of cause, size, location, or complexity, provides a common, nationwide approach to enable the whole community to work together to manage all threats and hazards.

This section explains the roles of the NRC, other Federal agencies, licensees, and State, Tribal and local governments during the response to an incident. It also explains the security issues associated with supporting the response efforts.

16.1.2.1 Federal Response

The Federal response structure was revamped after 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.

The DHS will assume overall Federal incident management coordination responsibilities when any one of the following three conditions applies:

(1) A Federal department or agency acting under its own authority has requested DHS assistance.

(2) The resources of State, Tribal and local authorities are overwhelmed, and the appropriate State, Tribal and local authorities have requested Federal assistance.

(3) The President of the United States has directed the Secretary to assume incident management responsibilities.

In 2008, 2011, 2013, and 2016, the governing documents outlining the responsibilities of the Secretary of Homeland Security, DHS, and other Federal, State, Tribal, and local entities were updated. These documents were related to NIMS and the National Response Framework, the 2016 Response Federal Interagency Operational Plan, and its associated annexes.

NIMS is a comprehensive, national approach to incident management that applies at all jurisdictional levels and across functional disciplines. NIMS enables Federal, State, Tribal, 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 in concert 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) and their associated Federal interagency operational plans provide guidance on Federal coordinating structures and processes to prevent, prepare for, mitigate, 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 Nuclear/Radiological Incident Annex to the Response and Recovery Federal Interagency Operational Plans, which describes the roles of lead Federal agencies with primary authority for response (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 (e.g., a terrorist-related incident), DHS is responsible for the overall domestic incident management, while the lead Federal agency coordinates the Federal on-scene actions and helps State, Tribal, and local governments determine measures to protect life, property, and the environment. The lead Federal agency will respond as part of the Federal response in accordance with the Nuclear/Radiological Incident Annex and with its own authorities. During incidents with offsite consequences, DHS may assume coordination of the Federal response, while the lead Federal agencies will continue to oversee the onsite response, monitor and support owner or operator activities (where applicable), 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, Tribal, and local government agencies on implementing protective actions. The lead Federal 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.

16.1.2.2 Licensee, State, Tribal, and Local Response

The NRC recognizes the nuclear power plant operator (licensee) and the State, Tribal, or local government as the two primary decisionmakers during a radiological incident at a licensed power reactor. The licensee is primarily responsible for the timely classification of an emergency; mitigating the consequences of an incident on site; and the prompt recommendation of protective actions to State, Tribal, and local authorities. The State, Tribal, or local governments are ultimately responsible for implementing appropriate protective actions for public health and safety.

16.1.2.3 NRC 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.

The NRC site team responds 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, which is usually established 48 to 72 hours after an incident). Through participation in these response centers, the NRC site team has access to wide-ranging State and Federal response assets, as well as to extensive radiological monitoring capabilities through DOE (i.e., field teams and aerial monitoring).

The NRC regularly participates in nuclear power plant and Federal interagency exercises each year to ensure its readiness to respond. The NRC also participates in the planning and conduct of the annual continuity of operations exercise each year and National Level Exercises every 2 years. The NRC's participation in such exercises gives the agency a valuable perspective on event response. This perspective improves interagency cooperation and imparts a better understanding of response roles during emergencies.

16.1.2.4 Aspects of Security that Support Response

Following the events of September 11, 2001, the NRC 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 access to substantial resources and as many as 18 Federal agencies available to help mitigate the radiological consequences of a serious accident or successful attack.

16.1.3 Implementation of Emergency Preparedness Measures

16.1.3.1 Emergency Classification System and Emergency Action Levels

Under 10 CFR 50.47(b)(4), 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. Section IV.C.1 of Appendix E to 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. Each of the four emergency classification levels is based on plant conditions (e.g., plant system status, in-plant and effluent radiological parameters, fission product barrier status, and other in-plant hazards) or external events (e.g., flooding, earthquakes, high winds, security events). These conditions form the basis for each licensee to establish specific thresholds and indicators, known collectively as emergency action levels for various plant conditions and external events.

Licensees and State, Tribal, and local agencies have established specific procedures for carrying out emergency plan actions 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.

The NRC has endorsed generic guidance documents that may be used to aid in the development of a licensee-specific emergency action level scheme. NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6, dated November 2012, provides the latest NRC-endorsed guidance for the development of emergency action levels. NEI 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, dated July 2009, contains NRC-endorsed guidance for developing emergency action levels for the AP1000 and the General Electric-Hitachi's economic simplified boiling water reactor (ESBWR) reactor designs. Additional guidance for developing a licensee-specific emergency action level scheme is listed on the NRC Web site at: https://www.nrc.gov/about-nrc/emerg-preparedness/about-emerg-preparedness/emerg-action-level-dev.html.

These documents are all considered generic guidance, as they are not plant-specific and may not be entirely applicable for some reactor designs (note that NEI 07-01 applies only to the AP1000 and ESBWR designs). However, the guidance 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, the applicant or licensee, and State and local governmental authorities must agree upon initial emergency action levels, and the NRC must approve these levels. Thereafter, the State and local governmental authorities must review emergency action levels annually. The NRC must approve any subsequent revision to an emergency action level scheme before implementation.

The nuclear industry developed severe accident management guidelines (SAMGs) in response to the Three Mile Island accident based on extensive research on severe accident phenomena. The purpose of SAMGs is to enhance the ability of plant operators to manage accident sequences that progress beyond emergency operating procedures and other applicable plant procedures. Following the Fukushima Dai-ichi accident, the nuclear industry and the NRC revisited the issue of SAMGs. In SRM-SECY-15-0065, “Proposed Rulemaking: Mitigation of Beyond Design Basis Events,” dated August 27, 2015, the Commission directed that SAMGs continue to be implemented voluntarily rather than being imposed as an NRC requirement. Following this direction, 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 decision, the NRC will provide periodic oversight of the SAMGs through the Reactor Oversight Process. Sections 12.2.3 and 19.4 of this report present additional information about emergency operating procedures.

16.1.3.2 Offsite Emergency Planning and Preparedness

The accident at Three Mile Island Nuclear Station, 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 in 10 CFR 50.33(g) and 10 CFR 50.54(s) 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 pathway emergency planning zone, as well as the plans of State Governments within the ingestion pathway zone.

In December 1979, the U.S. 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, “Memorandum of Understanding Between the Department of Homeland Security / Federal Emergency Management Agency and Nuclear Regulatory Commission Regarding Radiological Response, Planning and Preparedness,” subsequently established FEMA’s role and responsibilities.

FEMA provides its findings on the acceptability of the offsite radiological emergency plans and preparedness to the NRC, which has the ultimate authority 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. Consistent with 10 CFR 50.47(a), 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.

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 identifies evaluation criteria that outline 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 and States, Tribal, and local Governments to develop acceptable radiological 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 discussed in Sections 16.1.2.2 and 16.1.4 of this report.

Through the Steering Committee for Emergency Planning, established under the NRC-FEMA memorandum of understanding, both agencies discuss and coordinate 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 regulations and guidance related to radiological emergency preparedness.

16.1.3.3 Emergency Preparedness Facilities

In 10 CFR 50.47, the NRC requires that a power reactor licensee have and maintain adequate emergency facilities and equipment to support the emergency response. Emergency facilities include a licensee onsite technical support center, which provides plant management and technical support to reactor operating personnel in the control room; an onsite operational support center, which serves as an assembly area for licensee support personnel; and an emergency operations facility, which serves as a near-site support facility for the management of the overall licensee emergency response, including coordination with Federal, State, Tribal, and local officials. In addition, NRC regulations require that a physical location or locations are established in advance to coordinate dissemination of information to the public.

The U.S. nuclear power industry also has developed, maintained, and operated two national response centers, one in Memphis, TN, and a second in Phoenix, AZ. These centers are a component of the licensees’ FLEX strategies and are therefore equipped with portable backup generators, pumps, cables, and standardized couplings and hoses, which can be moved to any U.S. nuclear power plant within 24 hours of a request using ground or air transport. Equipment at the response centers supplements permanent safety systems built into nuclear energy facilities and multiple sets of portable, backup safety equipment already positioned at the facilities.

16.1.3.4 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 are included in NUREG-0654/FEMA-REP-1, Revision 1, Supplement 3, "Guidance for Protective Action Strategies," issued in November 2011, and EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," dated January 2017. Supplement 3 to NUREG-0654/FEMA-REP-1 was updated in 2011 to reflect recommendations for enhancing protective action strategies developed from analyses of a spectrum of scenarios for a core melt accident at a nuclear power plant. 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 synonymous with a "severe accident" as that term is used in U.S. nuclear power plant accident analyses. Supplement 3 to NUREG-0654/FEMA-REP-1 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. The NRC also finds that potassium iodide is a reasonable, prudent, and inexpensive supplement to evacuation and sheltering for the public in specific local conditions. In 2001, the NRC amended its regulation in 10 CFR 50.47(b)(10) for emergency planning associated with potassium iodide. This amendment requires that each State consider the prophylactic use of potassium iodide as appropriate. In EPA-400/R-17/001, EPA, in cooperation with the cognizant agencies, updated the FEMA Federal Policy Guidance on the Use of Potassium Iodide Prophylaxis.

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.

16.1.4 Emergency Response Exercises

The NRC and FEMA coordinate their evaluation of periodic emergency response exercises and require all operating nuclear power plant sites to conduct an exercise every 2 years, as outlined in Section IV.F.2.b of Appendix E to 10 CFR Part 50. These mandatory full-participation exercises are integrated efforts by the licensee and State, Tribal, and local radiological emergency response organizations that play a role the licensee's radiological 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 through 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 within the plume exposure pathway to participate in biennial exercises and the State to participate in an ingestion pathway exercise with a nuclear power plant located in the State every 8-year exercise cycle.

16.1.5 Regulatory Review and Inspection Practices

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 seven cornerstones of safety in the Reactor Oversight Process. 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—NRC 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—NRC inspectors evaluate drills and simulator-based training evolutions in which shift operating crews and licensee emergency response organization members participate.

- Exercise Evaluation—NRC 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 NRC inspectors assess whether the licensee's self-critique is consistent with their observations. The emergency preparedness performance indicators for drill and exercise performance rely on 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 licensee's drill and exercise program.

- Alert and Notification System Evaluation—NRC inspectors verify how well the testing program complies with program procedures.

- Emergency Action Level and Emergency Plan Changes—NRC 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—NRC 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—NRC inspectors verify that the data reported for the performance indicator values are valid.

While 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 assist the NRC in its regulatory review 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 its 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.

16.2 Communications Activities

16.2.1 Communications with Neighboring States and International Arrangements

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

Under its bilateral agreements with Canada and Mexico, the NRC will promptly notify and exchange information in an emergency that has the potential for transboundary effects. The "Memorandum of Understanding between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission [CNSC] for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters," was most recently renewed in 2017 for a period of 5 years. The NRC and the CNSC have a close bilateral relationship and conduct technical bilateral meetings at least annually. The CNSC observed the national-level exercise at the LaSalle County Station in August 2018, and the NRC observed a CNSC exercise at the

Point Lepreau Nuclear Generating Station in October 2018. The “Arrangement between the United States Nuclear Regulatory Commission and the National Nuclear Safety and Safeguards Commission of the United Mexican States [CNSNS] for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters” was most recently renewed in 2017 for a period of 5 years. Arrangements are also being made with CNSNS to observe emergency response exercises in the United States and in Mexico in 2019.

The NRC also routinely practices emergency communications with 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. The NRC also hosts several bilateral exchanges every year about emergency preparedness and response activities and emergency exercise observation with foreign regulatory bodies at the NRC’s Headquarters in Rockville, MD, and at U.S. nuclear power plants around the country.

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 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 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 about the emergency event.

16.2.2 Communications with the Public

The emergency planning standard outlined in 10 CFR 50.47(b)(7) requires U.S. nuclear power reactor licensees to make information periodically available to the public on how it would be notified and what its initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors). The standard also requires 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 be established for coordinated dissemination of information to the public. The emergency planning standard outlined in 10 CFR 50.47(b)(5) also requires, in part, that the content of initial and followup messages to the public has been decided 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. Sections II.E and II.G of NUREG-0654/FEMA-REP-1 outline the evaluation criteria that provide an acceptable means for complying with the requirements of these emergency planning standards.

Section IV.D of Appendix E to 10 CFR Part 50 describes licensee requirements for promptly notifying the public of a declared emergency. The appendix also describes the yearly dissemination of basic emergency planning information to the public located within the plume exposure pathway emergency planning zone. That information includes the following:

- 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 list of local broadcast stations that would 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 to licensees and 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 regulations and guidance related to radiological emergency preparedness, and (3) using the NRC Web site, social media, and periodic newsletters for outreach.

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.

17.1 Background

The NRC's siting responsibilities stem from the Atomic Energy Act and the Energy Reorganization 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. Also, under the National Environmental Policy Act, which prescribes procedures for environmental reviews of Federal projects, the NRC evaluates the environmental impacts of siting a nuclear facility.

As discussed in Article 7 of this report, in 1989, the NRC developed 10 CFR Part 52 as an alternative regulatory approach to licensing 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. The NRC's defense-in-depth safety philosophy has, and will continue to, take into account the presence of densely-populated areas and the impact of population density on the effectiveness of emergency response actions. 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 impacts to the human environment during the construction and operation of the plant are appropriately considered.

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 non-seismic, 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," of 10 CFR Part 100, "Reactor Site Criteria." License applicants must consider the siting factors specified in 10 CFR 100.20, "Factors To Be Considered When Evaluating Sites," that include population distributions, proximity to man-made hazards, and the physical characteristics of the proposed site. The criteria in 10 CFR 100.21, "Non-seismic Siting Criteria," restrict occupancy around the site and establish limits on radiological releases and dose consequences from normal operations and postulated accidents. Additionally, 10 CFR 100.23, "Geologic and Seismic Siting Criteria," requires evaluation of all factors that might affect the design and operation of the proposed facility and establishes 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. Subsequent sections under Article 17 of this report discuss the specifics of applicable RGs. RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 2, dated April 1998, provides a general set of safety and environmental criteria that the NRC staff has found useful in assessing candidate site identification in specific licensing cases. NUREG-0800 guides the staff in reviewing the site safety content of the applicant's safety analysis report. RS-002, "Processing Applications for Early Site Permits," dated May 3, 2004, identifies parts of NUREG-0800 that apply to the review of early site permits.

17.2.2 Assessments of Non-seismic Aspects of Siting

Siting facilities away from densely populated areas is a principal component of the 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 projectiles, 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, dated December 2001; RG 1.91, “Evaluations of Explosions Postulated To Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,” Revision 2, dated April 2013; and RG 1.217, “Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts,” Revision 0, dated 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.” RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” Revision 1, dated March 2013, gives acceptable approaches for obtaining meteorological data. 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, dated March 2007, and RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” Revision 0, dated 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. RG 1.27, “Ultimate Heat Sink for Nuclear Power Plants,” Revision 2, dated January 1976, addresses considerations for water supply. Because of the likely proximity to water, many sites need to be evaluated for flood hazards from precipitation, wind, tsunami, or those related to man-made hazards such as dam failures. RG 1.59, “Design Basis Floods for Nuclear Power Plants,” Revision 2, dated August 1977, provides acceptable approaches for conducting flood-hazard evaluations.

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 an emergency. In addition, 10 CFR 100.21 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 approach described in RG 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion," dated 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 the risk-informed, performance-based ground motion response spectrum 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," dated January 2012), which built on previous seismic source models. The new seismic source model used a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 assessment process to represent the center, body, and range of technically defensible interpretations of the available data, models, and methods. This model is described in NUREG-2213, "Updated Implementation Guidelines for SSHAC Hazard Studies," dated October 2018. The updated model provides a consistent and stable basis for evaluating seismic source zones in probabilistic seismic hazard assessments for the central and eastern United States.

The NRC reviews and certifies new and advanced reactor designs under 10 CFR Part 52. The seismic capacity of the certified designs is determined independent of any specific site; however, the design is intended to be capable of being located in most currently existing sites. Because a seismic probabilistic risk assessment requires site-specific hazards information, the NRC requires a seismic margin analysis for all new and advanced reactor designs. This analysis evaluates the sequence-level ability of plant SSCs to withstand an earthquake with high confidence (i.e., 95 percent) of low probability (i.e., 5 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," 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 more 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 not pose 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 SSCs. Commitments (including the radiological acceptance criteria) the applicant made during siting and documented in its final safety analysis report remain binding until modified. A licensee must evaluate the potential consequences of design changes against these radiological criteria to demonstrate that the changes will result in a design that still complies with the regulations and commitments. If the consequences increase more than minimally, as outlined in 10 CFR 50.59, or require a change to the technical specifications, as discussed in Article 14 of this report, the licensee must obtain NRC approval before implementing the proposed modification. Requirements in 10 CFR 50.67 allow licensees to use an alternative source term in place of the accident source term used in the original licensing and siting of the operating facility.

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, ESBWR, and APR1400 designs with similar results and will apply it to all contemplated light-water reactor design certification application reviews, including the Mitsubishi U.S. APWR, and NuScale. For nonlight-water reactor designs and advanced reactors, applicants will have to describe their rationale for an appropriate accident source term characterization, which 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 with regulatory margin. Since the issuance of 10 CFR 50.67 in 1999, most operating reactor licensees have 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 impacts of siting consist of the plant's demands on the environment (e.g., water use and effects of construction and operation). These impacts are addressed in 10 CFR Part 51, which implements the National Environmental Policy Act consistent with the NRC's statutory authority and reflects the agency's policy of voluntarily applying the regulations of the President's Council on Environmental Quality, subject to certain conditions. The NRC considers environmental impacts 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. RG 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," Revision 3, dated September 2018, guides applicants in preparing environmental reports (which the agency uses to prepare the environmental impact statement) for a range of applications, including site reviews for construction permits and operating licenses under 10 CFR Part 50 and for early site permits and combined licenses under 10 CFR Part 52. The environmental standard review plans contain the NRC staff guidance for environmental reviews for the above applications. The NRC is in the process of updating NUREG-1555, the environmental standard review plan, to align with the updated guidance in RG 4.2, Revision 3.

Environmental standard review plans are also applicable to environmental reviews of applications for combined licenses under 10 CFR Part 52, Subpart C, "Combined Licenses," when the applications reference an early site permit. Reviews of early site permit applications are limited in scope because the reviews focus on the environmental effects of nuclear power plant construction and operation with 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, Revision 1, to NUREG-1555 guide the staff's environmental review for license renewal applications under 10 CFR Part 54. Article 14 of this report discusses the license renewal process in more detail.

Several other NRC actions on siting and site suitability require environmental reviews, including issuance of limited work authorizations under 10 CFR 50.10(e); 10 CFR 52.25, "Extent of Activities Permitted"; and 10 CFR 52.91, "Authorization To Conduct Limited Work Authorization Activities", early partial decisions under 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 Construction Permit or Combined License to Construct

Certain Utilization Facilities; and Advance Issuance of Limited Work Authorizations," of 10 CFR Part 2, and preapplication reviews of site suitability issues under Appendix Q, "Pre-Application Early Review of Site Suitability Issues," to 10 CFR Part 50.

17.3.2 Other Considerations for Environmental Reviews

The NRC's first environmental standard review plan was 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, approving sites independent of these designs, and then efficiently linking these approvals to approve construction and operation of the facility. The NRC has approved five early site permits and 14 combined license applications under 10 CFR Part 52 and is reviewing one additional early site permit.

As part of the revisions to the licensing framework, the NRC issued RS-002, which incorporates the environmental guidance in the environmental standard review plan, NUREG-1555, and the outcome of interactions with stakeholders. 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, in June 2007, the NRC issued RG 1.206, "Combined License Applications for Nuclear Power Plants," which includes guidance on the assessment of environmental issues. In October 2018, the NRC issued RG 1.206, Revision 1, "Applications for Nuclear Power Plants." This revision reflects lessons learned from the review of large light-water nuclear power plant applications under 10 CFR Part 52, since the initial issuance of RG 1.206 in June 2007.

In September 2014, the NRC issued a revision to 10 CFR 51.23, "Environmental Impacts of Continued Storage of Spent Nuclear Fuel beyond the Licensed Life for Operation of a Reactor," 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 about the environmental impacts of continued storage of spent nuclear fuel beyond a reactor's operating license.

17.4 Reevaluation 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 or an action effecting issue finality under 10 CFR Part 52 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, including 10 CFR 50.54(f), to request that licensees reevaluate the seismic and flooding hazards at their sites using current regulatory guidance and methodologies and, if necessary, perform a risk evaluation. All licensees have completed these seismic and flooding reevaluations, while some risk evaluations are ongoing. The results of these risk evaluations, where applicable, are being used to determine whether additional regulatory actions are necessary to ensure that plants are adequately protected from seismic and flooding events. Section 2.3.3.4 of this report provides additional information on the NRC's implementation of Fukushima lessons learned.

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 can 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.

The NRC's Tribal Policy Statement was published on January 9, 2017 (82 FR 2402). The Tribal Policy Statement establishes principles to be followed by the NRC staff to promote effective government-to-government interactions with American Indian and Alaska Native Tribes. This policy statement applies to all NRC interactions with Tribes including siting new reactors.

17.6 Vienna Declaration on Nuclear Safety

On February 18, 2015, the contracting parties to the CNS issued the Vienna Declaration on Nuclear Safety in INFCIRC 872. The declaration does not establish new requirements but recommits the contracting parties to the implementation of the CNS principles and objectives to prevent accidents and mitigate radiological consequences, as discussed in Articles 6, 14, 17, 18, and 19. Section 2.4.1.2 of this report summarizes the United States' implementation of these CNS objectives.

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.

18.1 Implementation of Defense-in-Depth

This section explains the defense-in-depth philosophy followed in regulatory practice, governing documents, and regulatory process for designing, constructing, and operating a nuclear power plant. It also discusses relevant experience and examples.

18.1.1 Overview of Regulatory Requirements and Governing Documents

Defense-in-depth is essential to a regulatory structure designed to provide for adequate protection of the public health and safety. Below is a list of important regulatory requirements and governing documents.

- Appendix A and Appendix B to 10 CFR Part 50

- NUREG/KM-0009, “Historical Review and Observations of Defense-in-Depth,” dated April 2016

- SECY-13-0132, “U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report,” dated December 6, 2013

- SRM-SECY-13-0132, “Staff Requirements—SECY-13-0132—U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report,” dated May 19, 2014

- NUREG/CR-6303, “Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems,” dated December 1994

- EPRI Technical Report 1002835, “Guideline for Performing Defense-in-Depth and Diversity Assessments for Digital Upgrades,” dated December 2004

- DI&C-ISG-02, Revision 2, “Task Working Group #2: Diversity and Defense-in-Depth Issues,” dated June 5, 2009

- Branch Technical Position 7-19, “Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems,” Revision 7, dated August 2016

- RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, dated January 2018

18.1.2 Application of the Defense-in-Depth Philosophy

Defense-in-depth is an element of the NRC’s safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena that, because they are unknown or unforeseen, are not reflected in either the PRA or traditional engineering analyses. The SRM on SECY-98-144, “Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation,” dated March 1, 1999, provides additional information on defense-in-depth as an element of the NRC’s safety philosophy.

In addition, nuclear plants that leverage the defense-in-depth philosophy in the design of the plant can gain some flexibility in operations and maintenance. For example, testing and maintenance of SSCs or corrective action to restore an engineered safety system might be allowed for short periods while remaining at-power consistent with established technical specifications. The NRC recognizes and allows these temporary configurations within these established programs. If a licensee proposes a licensing basis change that permits new or extended entry into a temporary condition, the NRC’s guidance states that the licensee should demonstrate that entry into that temporary condition is justified and that consistency with the defense-in-depth philosophy is maintained as described in this section.

Defense-in-depth is often characterized by varying layers of defense, each of which may represent conceptual attributes of nuclear power plant design and operation or tangible objects such as the physical barriers between fission products and the environment. For power reactors, the NRC typically treats defense-in-depth as four layers of defense that are a mixture of conceptual constructs and physical barriers (see RG 1.174, Revision 3, for further detail):

- robust plant design to survive hazards and minimize challenges that could result in an event occurring,

- prevention of a severe accident (core damage) if an event occurs,
- containment of the source term if a severe accident occurs, and
- protection of the public from any releases of radioactive material (e.g., through siting in low-population areas and the ability to shelter or evacuate people, if necessary).

18.1.3 Regulatory Review and Control Activities

Current applications to build new nuclear power plants have been submitted using the combined license process under 10 CFR Part 52, which is discussed in Article 19 of this report. Under 10 CFR Part 52, the applicant must submit a final safety analysis report for NRC review before authorization is granted to begin construction. The final safety analysis report is the principal document 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. This report should give the details of the final design of the facility, plans for operation, and procedures for coping with emergencies. 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.

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 the 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.

The NRC oversees nuclear power plant construction to ensure compliance with the agency's regulations. The NRC has developed an inspection program for nuclear plants licensed under 10 CFR Part 52 that incorporates inspections, tests, analyses, and acceptance criteria (ITAAC) from 10 CFR Part 52, as well as lessons learned from the 10 CFR Part 50 inspection program used in the previous construction era (1970-1980) and from the construction of Watts Bar Nuclear Plant, Unit 2. During construction, NRC inspectors review 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 use of ITAAC and the use of an adequate quality assurance program are two examples of defense-in-depth applied to the early stages of plant construction.

From the partial construction of Virgil C. Summer Nuclear Station, Units 2 and 3, and the ongoing construction of Vogtle Electric Generating Plant, Units 3 and 4, the NRC has learned that some ITAAC may not serve their original purpose and others need clarification to ensure proper completion. The clarification often requires a license amendment. The NRC and the nuclear industry worked to improve and standardize ITAAC wording by focusing on critical safety parameters.

In addition to inspections of ITAAC related work, the NRC inspection program addresses 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.

The NRC also interacts with manufacturers and suppliers of safety-related components through the NRC vendor inspection program that covers compliance with quality assurance and defect reporting requirements. The NRC conducts vendor inspections at vendor shops principally to examine whether the vendor has been complying with Appendix B to 10 CFR Part 50 and 10 CFR Part 21.

18.1.4 Experience and Implementation of Defense-in-Depth Measures

The NRC’s response to the Fukushima Dai-ichi accident demonstrates how the staff applied the defense-in-depth philosophy to address and evaluate the lessons learned from that event. The staff’s review of the Fukushima Near-Term Task Force recommendations identified areas for further evaluation to enhance the regulations and cope with events beyond the current design basis. 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. Because of the beyond-design-basis severity of the earthquake and tsunami that led to the accident at Fukushima Dai-chi, the NRC issued requests for information asking licensees to reevaluate their seismic and flooding hazards. The information obtained helped the NRC consider the protection levels for those events and determine whether additional regulatory action was needed.

The NRC also considered the need for the mitigation of beyond-design-basis external events and sought to build on prior defense-in-depth enhancements. As illustrated by the events at Fukushima Dai-ichi, the loss of electrical power can be an important contributor to the risk of nuclear power plant accidents. The NRC recognized this in NUREG-75/014 and addressed it in 10 CFR 50.63, “Loss of all Alternating Current,” (the Station Blackout Rule), in 1988. The conditions experienced at the Fukushima Dai-ichi nuclear power plant exceeded the conditions and time period of a station blackout in 10 CFR 50.63. Following the terrorist events of September 11, 2001, the NRC updated its regulations (10 CFR 50.54(hh)(2)) to require licensees to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under circumstances associated with loss of large areas of the plant as the result of explosions or fire. While these strategies were founded on the concept that an explosion or fire could challenge a plant’s key safety functions, they provide preplanned responses that could allow a licensee to respond to challenges to maintaining or restoring core cooling, containment, and SFP capabilities posed by natural hazards such as those at Fukushima Dai-ichi.

Following the Fukushima accident and the publishing of the Near-Term Task Force report, the U.S. nuclear industry proposed the FLEX initiative to develop an integrated safety-focused approach to expedite implementation of Fukushima lessons learned. The FLEX strategies focus on maintaining or restoring key plant safety functions and are not tied to any specific damage state or mechanistic assessment of external events. The NRC took this proposal into account, and on March 12, 2012, issued Order EA-12-049 requiring a three-phased approach for mitigating beyond-design-basis external events to maintain or restore key safety functions. This order expands on similar strategies required by 10 CFR 50.54(hh)(2).

FLEX provides an additional layer of defense by providing supplemental capabilities and strategies for responding to beyond-design-basis scenarios affecting all units at a site. The three-phased approach also applies the defense-in-depth philosophy. The FLEX strategies consist of an onsite component (using plant equipment followed by FLEX equipment stored at or near the plant site) and an offsite component (using additional materials and equipment for a longer-term) in responding to an accident and ensuring equipment availability and redundancy. The U.S. nuclear industry has established two national response centers to store and maintain the necessary offsite equipment, each capable of responding to any of the U.S. nuclear power plant sites, and multiple means to deliver the equipment and supplies to the sites. Additionally, in response to the NRC's March 12, 2012, information request, licensees assessed their emergency planning communication capabilities and staffing levels to confirm or enhance their capabilities to implement their mitigation strategies and respond to a beyond-design-basis event that affects all units at the site.

Section 2.3.3.4 of this report discusses in more detail the actions resulting from the Near-Term Task Force recommendations and the resulting orders and information request.

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 of all three.

For example, 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. Also, in its application for the APR1400, Korea Hydro and Nuclear Power submitted a topical report describing the safety injection tank fluidic device. The applicant stated that incorporation of the device into the APR1400 design, coupled with the LOCA mitigation strategy, simplifies an important safety system by integrating an inherently reliable passive safety component with the conventional safety injection system. This design improvement, in addition to the direct vessel injection, contributes to the acceptability of the elimination of low pressure safety injection pumps in APR1400s. The use of this device is also expected to reduce the maintenance and testing workload at nuclear facilities while maintaining a very high level of safety. The applicant provided the results of its full-scale testing. The test results, combined with the analyses and the LOCA mitigation strategy, were enough to demonstrate that the device will perform as stated.

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

The NRC uses NUREG-0800, Chapter 18, Revision 3, to support its reviews of the human factors engineering issues associated with the certification and licensing of new plant designs. Chapter 18 is supported by NUREG-0700, Revision 2, for reviewing human factors aspects of human-system interface designs. In November 2012, the NRC issued NUREG-0711, Revision 3, to support the review of human factors design programs. NUREG-0800, Section 14.3.9, “Human Factors Engineering—Inspections, Tests, Analyses, and Acceptance Criteria,” dated March 2007, provides guidance for human factors ITAAC inspections. Section C.I.18 of RG 1.206 addresses the human factors engineering review of combined license applications.

18.3.2 Experience

The NRC is actively reviewing standard designs for NuScale and the U.S. APWR design certifications and recently completed review of the APR1400 standard design. Currently, no combined license applications are under review.

18.3.2.1 Human Factors Engineering

The NRC’s human factors engineering reviews for design certification applications focus on evaluating implementation plans for the design of the control facilities to ensure that the design process will be carried out consistently 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 staff recently conducted oversight of integrated system validation testing (as well as other elements of human factors programs described in NUREG-0711). The integrated system validation provides performance-based evidence that the design can be used to safely control the plant. In 2018, the staff completed a series of audits of the NuScale integrated system validation testing and multiple ITAAC inspections of the AP1000 integrated system validation process. The staff also conducted the human factors program reviews of the APR1400 and NuScale applications.

18.3.2.2 Digital Instrumentation and Controls

Chapter 7, “Instrumentation and Controls,” of NUREG-0800 provides guidance to the NRC staff in reviewing the instrumentation and control design of 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 adequately protects public health and safety. All the new reactor designs contain highly integrated digital instrumentation and control systems, which have advantages but can also present issues that are not relevant to analog systems. Examples of these issues include the following:

- A common-cause failure attributable to software errors was not possible in analog systems. This possible failure mode may be addressed using 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 cybersecurity.

- Highly integrated control room designs with safety and nonsafety displays and controls are the norm for new reactor designs. Human factors design and quality assurance during all phases of software development, control, and validation and verification are critical.

The NRC developed several ISG documents for review of new and innovative digital instrumentation and control systems found in new reactor designs. The guidance also provided

the industry with the expectations and criteria the staff uses to evaluate designs and determine compliance with NRC regulations. The staff has been using this guidance, along with other existing sources such as NUREG-0800, in its review of applications for design certifications and combined licenses. The staff has incorporated some of the ISG into formal NRC staff guidance in NUREG-0800 and associated RGs. All ISG 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, ABWR, and APR1400 reactor designs as well as those for the Fermi, Unit 3 combined license. The staff is in the process of reviewing the instrumentation and control design for the U.S. APWR design certification, the NuScale small modular reactor design certification, and the ABWR design certification renewal. The staff continues to support the instrumentation and control ITAAC inspection activities for the AP1000 combined licenses and the AP1000 license amendment requests. The staff also continues to review digital instrumentation and control platform topical reports that can be referenced in subsequent site-specific license amendment requests.

To support the review of applications for small modular reactor design certifications and combined licenses, the NRC staff developed design-specific review standards. The guidance in the design-specific review standards modifies the guidance in the corresponding chapters of NUREG-0800 to reflect lessons learned from using NUREG-0800 to review new large light water reactor designs. For the design-specific review standard chapter on instrumentation and controls, the staff has incorporated the lessons learned into this guidance with the following goals:

- 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 significant aspects of the instrumentation and control design in a unified manner through this framework.

- To reflect an integrated instrumentation and control design using digital technology, which is common in new and advanced reactor designs. In addition, the topics 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.

- To 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.

- To address various new sources, such as the Multinational Design Evaluation Program and NEA’s common positions, and lessons learned from other countries.

- To 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 development and use of the design-specific review standard for the NuScale small modular reactor design have increased the efficiency and effectiveness of the instrument and control licensing review. The restructured, safety-focused approach in the design-specific review standard, which emphasizes the fundamental instrumentation and control design principles mentioned above, is a step forward for other future new and advanced reactor licensing applications.

The NRC participates in NEA's Committee on Nuclear Regulatory Activities, an international assembly of nuclear regulators and technical support organizations addressing common issues with the licensing of operating and new reactors. Specifically, the NRC chairs the Working Group on Digital Instrumentation and Control, which is looking at ways to harmonize requirements, standards, and guidance for instrumentation and control. The working group allows the NRC to share digital instrumentation and control information to support regulatory infrastructure improvements and licensing decisions. The NRC is also working with the AP1000 instrumentation and control technical expert subgroup under the Multinational Design Evaluation Program, which is an international collaboration of regulatory agencies engaged in reviewing the AP1000 instrumentation and control design.

Section 2.3.2.5 of this report discusses the digital instrumentation and control system regulatory program and processes.

18.3.2.3 Cybersecurity

After September 11, 2001, the NRC issued two security-related orders that required immediate identification and assessment of computer-based systems deemed critical to the operation and security of the facility.

Subsequently, in March 2009, the NRC issued a new rule on cybersecurity, 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks." The cybersecurity rule requires 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 cybersecurity rule requirements, operating power reactor licensees had to submit a cybersecurity plan, including a proposed implementation schedule with interim milestones, to the NRC for review and approval by November 23, 2009, and operating license 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 cybersecurity plan also must acknowledge a commitment to maintain the cybersecurity program and provide adequate documentation of how that will be accomplished.

In 2010, the NRC and the North American Electric Reliability Corporation entered into a 5–year memorandum of understanding to address nuclear plant cybersecurity roles, responsibilities, and areas of coordination between the two organizations. The NRC and the North American Electric Reliability Corporation renewed the 5-year memorandum of understanding in 2015.

After this memorandum of understanding, 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 also found the NRC's regulatory framework sufficient to meet the North American Electric Reliability Corporation's cybersecurity requirements for power generation plants. Under the memorandum of understanding, the NRC 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 two organizations renewed this 5-year memorandum of agreement 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. Stakeholders, including members of industry and representatives from DHS, the Federal Energy Regulatory Commission, and the National Institute of Standards and Technology, collaborated with the NRC in developing this program. The NRC completed inspection activities related to the interim milestones in calendar year 2015. Most NRC licensees implemented the remaining aspects of the program, including controls for a greater number of systems and processes, in 2017. The NRC started full implementation inspection activities in calendar year 2017 and will continue the inspections through 2020.

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 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 Program to support new reactor construction activities. To achieve this goal, the NRC staff developed a risk-informed process to collect, screen, evaluate, and apply construction experience insights to its new reactor licensing and construction oversight activities. In 2012, the NRC formed a center of expertise for Operating Experience, integrating the Construction Experience Program and the Operating Experience Program to increase efficiency and effectiveness. The agency completed this integration in 2016, resulting in the program described in Sections 6.3.5 and 19.7 of this report.

The review of operating experience routinely examines and evaluates the potential impact of issues, including issues at operating reactors that could provide potential lessons learned for new reactor construction. This includes events related to latent design and construction deficiencies, significant design changes, installation and testing activities, and heavy loads.

Section 19.7 of this report describes the followup to actions related to evaluation, communication, and application of construction experience. This includes the use and sharing of new construction experience with international counterparts, which was formerly done via NEA's Construction Experience Program. The contents of the database associated with this NEA program are being relocated to IAEA's International Reporting System for Operating Experience database.

18.5 Vienna Declaration on Nuclear Safety

On February 18, 2015, the contracting parties to the CNS issued the Vienna Declaration on Nuclear Safety in INFCIRC 872. The declaration does not establish new requirements but recommits the contracting parties to the implementation of the CNS principles and objectives to prevent accidents and mitigate radiological consequences, as discussed in Articles 6, 14, 17, 18, and 19. Section 2.4.1.2 of this report summarizes the United States' implementation of these CNS objectives.

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, tests, 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 commercial nuclear facility and in monitoring its safe operation throughout its service life. This section describes the most significant regulations and programs corresponding to each obligation of Article 19.

19.1 Initial Authorization to Operate

In the United States there are two processes for requesting permission to construct and operate a nuclear power plant. Both require NRC approval.

All currently operating reactors in the United States received licenses under the two-step process in 10 CFR Part 50. This licensing process requires both a construction permit and an operating license. In the operating license process, a public hearing is neither mandatory nor automatic. However, soon after the NRC accepts the application for review, it publishes a notice in the *Federal Register* stating that it has received the application, had accepted it for review, and is considering issuance of the license. This notice states that any person whose interest might be affected by the proceeding may petition the NRC for a hearing. The Atomic Safety and Licensing Board will determine whether to grant or deny the request for a hearing. The Advisory Committee on Reactor Safeguards will conduct an independent safety review and report to the Commission.

The additional licensing processes in 10 CFR Part 52 provide for site approvals and design approvals in advance of construction authorization. In addition, 10 CFR Part 52 includes a process that combines a construction permit and an operating license with conditions into one license (a combined license) for a nuclear power plant. The NRC must hold a public hearing (uncontested hearing) before it issues a construction permit, early site permit, or combined license. 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 a contested hearing.

An early site permit issued under Subpart A, “Early Site Permits,” of 10 CFR Part 52, provides for resolution of site safety, environmental, 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 issues in an environmental impact statement. The early site permit may also allow limited construction activities under 10 CFR 50.10, “License Required; Limited Work Authorization,” subject to redress, during the review of a combined license. After its review, 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. 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. The staff has not approved any new early site permits since the issuance of the last U.S. National Report. One new early site permit application for the Clinch River Nuclear Site is under review.

The NRC also may certify a standard plant design through a rulemaking under Subpart B, “Standard Design Certifications,” of 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 six standard plant designs under the design certification process: (1) General Electric’s ABWR, (2) Westinghouse Electric Company’s System 80+ (originally designed by Combustion Engineering), (3) Westinghouse’s AP600 design, (4) Westinghouse’s AP1000, (5) General Electric-Hitachi’s ESBWR, and most recently, in May 2019, (6) Korea Hydro and Nuclear Power’s APR1400. Two applications are currently under review: (1) Mitsubishi’s U.S. APWR and (2) NuScale Power’s small modular reactor. In addition, the NRC staff has received two applications to renew the ABWR design certification; the NRC is actively reviewing the General Electric Hitachi ABWR renewal application.

A combined license, issued under Subpart C, “Combined Licenses,” of 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. Like 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 four combined licenses authorizing construction and operation of new nuclear power plants at two sites in the United States. To date, the NRC has issued 14 licenses at eight sites. Currently, eight licensees at five sites remain in place; the others were terminated at the licensees' request. There are no combined license applications currently under review.

After issuing a combined license, the NRC staff will verify that the licensee has performed the required 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 acceptance criteria. However, requests for such a hearing will be considered only if the petitioner shows 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 are 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 and evaluations in the safety analysis report, tests, and operational experience. The regulations 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. The technical specifications must contain (1) safety limits and limiting safety system settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. Licensees cannot change the technical specifications without prior NRC approval.

The NRC maintains vendor-specific standard technical specifications in NUREG-1430 through NUREG-1434 and NUREG-2194, "Standard Technical Specifications for Westinghouse Advanced Passive 1000 (AP1000) Plants," Volumes 1 and 2, dated April 2016.

The NRC encourages licensees to use the standard technical specifications as the basis for plant-specific technical specifications. The agency also considers requests to adopt parts of the 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 have developed risk-informed improvements to technical specifications. Recently, the NRC approved a technical specifications program allowing licensees to determine the appropriate surveillance test intervals based in part on risk information. The agency approved another technical specifications program allowing licensees an option to determine the appropriate out-of-service times for equipment, based in part on the risk profile of the overall plant configuration. These optional improvements allow operational flexibility while maintaining or improving safety, reducing unnecessary burden and making technical specifications congruent with the agency's other risk-informed regulatory requirements.

19.3 Approved Procedures

In the United States, operations, maintenance, inspection, and testing of a commercial nuclear facility 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 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," dated November 1980; NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," dated January 1983; and NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," dated August 1982.

After the 1979 accident at Three Mile Island Nuclear Station, 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, "Operating and Emergency Operating Procedures."

Furthermore, following the Fukushima accident, the NRC ordered all power reactor licensees to develop mitigation strategies to respond to beyond-design-basis events affecting all units at a site for an indefinite period of time. Section 2.3.3.4 of this report discusses this in more detail. FLEX support guidelines are used to implement the strategies developed in response to Order EA-12-049. The industry guidance for complying with this order provides a procedural approach for the implementation of FLEX strategies, which includes evaluating these strategies for integration with the existing procedures, including emergency operating procedures. All operating U.S. power reactors have completed the required safety enhancements and have reported their compliance with Order EA-12-049. The NRC staff has reviewed the licensees' required plans and strategies and has completed onsite inspections to confirm each licensee's implementation of the order. A final rule, 10 CFR 50.155, was approved by the Commission making the requirements of the mitigation strategies order generically applicable in the NRC's regulations.

In SRM-SECY-15-0065, the Commission directed that SAMGs continue to be implemented voluntarily rather than being imposed as an NRC requirement. In response, 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. Sections 12.2.3 and 16.1.3.1 of this report provide additional information on emergency operating procedures and emergency classification levels.

19.5 Availability of Engineering and Technical Support

In 10 CFR 50.120, the NRC requires operating license applicants and combined license holders 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 prior to fuel loading either by confirming that the licensee's training programs have been accredited by the National Nuclear Accrediting Board or by performing an inspection of the training programs if they have not been accredited. In addition, 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. Equipment performance may provide insights into the availability of trained and competent engineers. 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, and NRC 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, which may include the availability of engineering and technical support, 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 incorporating 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 LERs. All 10 CFR 50.72 event notifications and 10 CFR 50.73 LERs, except those containing sensitive security-related information, are available on the NRC's public Web site at https://www.nrc.gov/reading rm/doc collections/event status/. The NRC is currently considering a petition for rulemaking to change the reporting requirements of 10 CFR 50.72. Additional information can be found at https://www.regulations.gov (Docket ID: NRC-2018-0201).

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 operating experience, monitor performance, identify precursors of more significant events, and provide operating 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 the Commission determines to be significant from a standpoint of public health and safety, is also publicly available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0090/.

NUREG-1022, “Event Reporting Guidelines: 10 CFR 50.72 and 50.73,” last updated with Supplement 1, “Event Report Guidelines 10 CFR 50.72(b)(3)(xiii),” in September 2014, 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 examples and definitions of key terms and phrases.

Event reporting under these rules, which were first issued in 1983, 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, event reporting data has reflected improvements in reactor safety system performance and decreasing trends in the number of reactor transients and significant events. For example, between 2007 and 2018, there were only two U.S. reactor events that were significant from a standpoint of public health and safety, as defined by the abnormal occurrence criteria in NUREG-0090:

- On June 7, 2011, at Fort Calhoun Station, an improperly replaced electrical breaker resulted in a fire which affected safety-related equipment

- On October 23, 2010, at Browns Ferry Nuclear Plant, Unit 1, a failure to meet residual heat removal low pressure coolant injection flow control valve design requirements resulted in a valve disk to stem separation, loss of safe shutdown functions, and loss of fire mitigation capabilities

In addition, the NRC participates in international event reporting systems. The NRC reviews each reported 10 CFR 50.72 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,” dated January 14, 2002, and IN 2009-27, “Revised International Nuclear and Radiological Event Scale User’s Manual,” dated November 13, 2009.

19.7 Programs To Collect and Analyze Operating Experience

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 has established a center of expertise to integrate the Construction Experience Program (described in Section 18.4 of this report) into the Operating Experience Program. Since the completion of this integration in 2016, the description of the Operating Experience Program here is also broadly applicable to the review of construction experience.

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. These databases include event reports, international reports, and inspection findings. In 2010, the NRC began adding additional information into a broader database that provides the same type of centralized data storage and retrieval options for non-reportable 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 LERs; reports of defects and noncompliance submitted under 10 CFR Part 21; 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.

As outlined in GL 82-04, "Use of INPO SEE-IN [Significant Event Evaluation and Information Network] Program," dated March 9, 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 to communicate significant events to the industry. In addition, INPO's Consolidated Events System gives member utilities the ability to report lower level events and equipment failure data to the Institute. INPO shares this data with all its members and, in a limited fashion, with the NRC.

Items that do not require significant evaluation are still reviewed and considered by the NRC staff for followup actions. These items can include e-mail 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 inspection staff may be summarized for consideration in the annual inspection planning and assessment reviews. 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. This 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 also participates in the International Nuclear and Radiological Event Scale and the IAEA international reporting system for operating experience both to communicate the safety significance of events, to share operating experience internationally, and to 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 IAEA's international reporting system for operating experience reports as they are posted. The NRC submits all relevant U.S. reactor-related generic communications to the IAEA international reporting system for communication to the international community along with selected LERs related to events that have attracted international interest.

19.8 <u>Radioactive Waste</u>

The NRC has issued regulations and guidance for nuclear power reactor licensees to ensure the safe management and disposal of low-level radioactive waste. Onsite low-level waste must be managed in accordance with the NRC regulations in 10 CFR Part 20 and 10 CFR Part 50. For example, Subpart K, "Waste Disposal," of 10 CFR Part 20, addresses 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 NRC last updated the low-level waste storage guidelines in RIS 2011-09, "Available Resources Associated with Extended Storage of Low-Level Waste," dated August 16, 2011.

Notwithstanding these regulations and guidance, the rising cost of waste disposal in the United States has encouraged practices to minimize the generation of radioactive waste. In June 2008, the NRC published RG 4.21 and on May 1, 2012, published the Policy Statement, "Low-Level Radioactive Waste Management and Volume Reduction" (77 FR 25760). The Policy Statement is a revision of the NRC's 1981 Policy Statement on "Low-Level Radioactive Waste Volume Reduction" (46 FR 51100) 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 30-60 cubic meters per unit) of operational waste each year. Radioactive wastes are treated as necessary to produce a structurally stable, final waste form and to minimize the release of radioactive and hazardous components to the environment. In 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," the NRC provides detailed regulations for designing and operating low-level waste disposal facilities. There are currently four low-level waste disposal facilities in the United States, all of which are regulated by Agreement States.

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[13] in 10 CFR Part 72.

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 Sixth National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management," Revision 5, dated October 2017) is available on the DOE Environmental Management Web site at https://www.energy.gov/sites/prod/files/2017/12/f46/10-20-17 6th _US_National_Report (Final).pdf.

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13 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.

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, the NRC completed the safety evaluation report for Yucca Mountain. In addition, the NRC developed a supplement to DOE's environmental impact statement to address ground water impacts previously identified by the NRC staff as requiring additional analysis.

In January 2015, the NRC staff found that DOE's license application met the regulatory requirements for the proposed repository except that DOE had not obtained certain land withdrawal and water rights necessary for construction and operation of the repository. 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

On February 18, 2015, the contracting parties to the CNS issued the Vienna Declaration on Nuclear Safety in INFCIRC 872. The declaration does not establish new requirements but recommits the contracting parties to the implementation of the CNS principles and objectives to prevent accidents and mitigate radiological consequences, as discussed in Articles 6, 14, 17, 18, and 19. Section 2.4.1.2 of this report summarizes the United States' implementation of these CNS objectives.

PART 3

The Role of the Institute of Nuclear Power Operations in Supporting the United States Commercial Nuclear Power Industry's Focus on Nuclear Safety

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 2019

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 the Three Mile Island Nuclear Station to promote the highest levels of safety and reliability—*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 22 members. In addition, many utility organizations that jointly own U.S. nuclear power plants are associate members. A number of major U.S. and international 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 (POs&Cs) and then by committing to meet these POs&Cs. 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. 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.

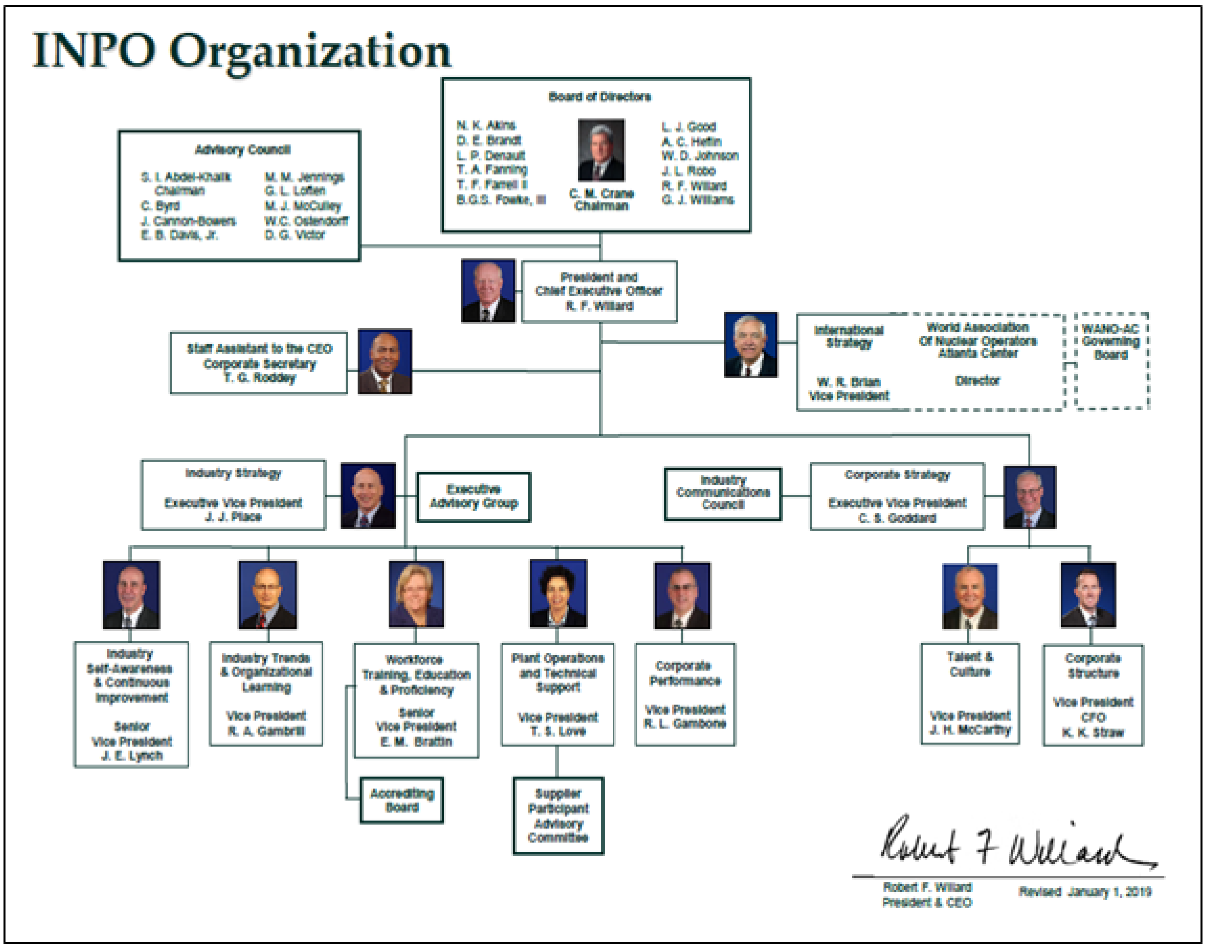
The nuclear industry’s commitment to go beyond regulatory compliance and continually strive for excellence, with INPO’s support, has resulted in substantial performance improvements over the past 40 years. At the end of 2018, the U.S. nuclear industry was performing at its highest levels ever. Today, the median industry capacity factor is above 93 percent, most plants experience no automatic scrams in a year, there were no significant operational events in 2018, and collective radiation dose and industrial accident rates are both lower by a factor of seven when compared with the rates of the 1980s industry. A focus for INPO going forward will be to help the industry sustain these high levels of performance.

Despite record levels of U.S. industry performance, challenges persist that warrant additional focus by the industry and INPO. For example, while the number of lower performing plants in the industry has significantly decreased, several remain, and others have not fully recovered to top-level performance. Progress has been made in reducing the number of consequential events in operations, maintenance, and engineering; however, more work is needed to further reduce errors in these areas that affect plant reliability. The number of plants operating with fuel defects has held steady over the last several years, and more emphasis is needed on reducing foreign material in primary systems in order to minimize the number of fuel failures. INPO is working closely with industry stakeholders to close these remaining performance gaps.

Numerous actions were taken in response to lessons learned from the March 2011 earthquake and tsunami in Japan that led to the consequential accident at Tokyo Electric Power Company's Fukushima Dai-ichi nuclear power plant. INPO conducts periodic reviews to ensure sustainability of those actions and to evaluate each member station's ability to respond to extreme external events.

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, composed of the INPO Chief Executive Officer (CEO) and 12 Chief Executives from INPO's member organizations, provides oversight of the Institute's operations and activities. The Institute's bylaws specify that at least two directors must have recent experience in the direct supervision of a nuclear power station. In addition, at least one director must represent a publicly-held utility. The president and CEO of the Institute, normally a single individual, is elected by, and reports to, the INPO Board of Directors. A chart depicting INPO's organization 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 important. The INPO Advisory Council of 9 to 15 professionals selected from outside INPO membership meets periodically to review the Institute's activities and to provide advice to both INPO management and the Board on its objectives and methods. Advisory Council members include prominent educators, scientists, engineers, business executives, and experts in organizational effectiveness, human relations, and finance.

INPO ensures that the industry actively participates in its programs and initiatives. Representatives from member utilities serve on an Executive Advisory Group, an Academy Council (assisting INPO's National Academy of Nuclear Training), and the Industry Communications Council. The Executive Advisory Group, composed of chief nuclear officers of all the member organizations, advises INPO management in nuclear technical areas as well as INPO operations. The Academy Council provides advice in the areas of training, accreditation, and human performance. The Industry Communications Council advises on the effectiveness of 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 in 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 increase their understanding of INPO's role and the importance of self-regulation across the industry.

- 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, utility insurance rates are 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 and monitoring processes, 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 the traditional corporate tasks of developing its workforce and maintaining and enhancing the infrastructure needed to support the way INPO does business:

- INPO is committed to developing and maintaining a diverse workforce and 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.

- 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>INPO's 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 five areas that challenge continuing improvement and sustainment of high levels of performance:

- **Self-awareness and continuous improvement**. 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.

- **Industry trends and organizational learning**. 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.

- **Workforce training, education, and proficiency**. 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.

- **Corporate performance**. Nuclear plants perform best when they are aligned with and receptive to oversight and support from a high performing corporate organization. In contrast, ineffective and misaligned corporate-plant engagement usually results, over time, in substandard plant performance and creates irregularities across fleets of nuclear stations. Similarly, a low performing plant will be impeded in its efforts to improve absent a strong corporate function. In addition, a low performing plant can adversely impact the effectiveness of corporate and the performance of other fleet stations. For the nuclear industry to achieve consistent, exemplary performance, utility corporate organizations must be uniformly high performing.

- **Fully recover low performers.** Corporate and site leadership are pivotal for recovering from low performance and in achieving workforce alignment and a culture of sustainability. They must have an unwavering focus on finding and fixing problems by being intrusive and engaged 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 identifying the means to augment leadership capability or capacity.

International Industry Strategy

Internationally, INPO leverages the World Association of Nuclear Operators (WANO) to influence the worldwide nuclear industry to improve nuclear safety and allow U.S. operators to benefit from worldwide operating experience.

- As a WANO member, INPO participates in peer reviews, member support missions, and other WANO activities worldwide and brings international participants to participate in similar INPO activities at U.S. stations.

- INPO serves as the collection point for U.S. nuclear station performance data and operating event information and shares this information with WANO, and, likewise, INPO receives international event information and disseminates it to the U.S. nuclear industry.

- Leveraging WANO's global reach, INPO liberally shares its intellectual property throughout the international industry.

- Through WANO, INPO provides services and products to support safe and reliable startup of new units by existing operators and new entrants. This includes a high level of engagement during construction and initial startup to instill superior standards among new entrants.

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

Financial and Human Resources

The 2019 operating budget for INPO of $108 million is primarily funded through member dues. Dues are approved annually by the INPO 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 330 full time employees 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 at INPO is approximately 400.

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, including performance monitoring, 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 2018, the industry provided INPO with nearly 900 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 public. This regulatory function is based principally on the Atomic Energy Act of 1954, as amended, and is carried out by the NRC. In 1979, following the accident at Three Mile Island, the President of the United States appointed a commission to investigate the accident. The Kemeny Commission, as it came to be known, 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 POs&Cs based on standards of excellence would effectively bring improvements. In the broadest sense, the ultimate goals of the NRC and INPO are the same in that both organizations strive to protect the public by promoting safe and reliable plant operations; therefore, some important areas of nuclear power plant operations are reviewed by both organizations. 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 —evaluator and regulator—may appear similar, they differ in some ways. The industry and INPO jointly develop numerous POs&Cs. INPO then conducts regular, extensive, and intrusive evaluations to determine how well they are being met. These POs&Cs are broad statements of conditions reflecting an excellent level of overall plant performance often exceeding regulatory requirements. These POs&Cs, 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 POs&Cs 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, continuous performance monitoring, and events analysis programs.

Nuclear operators are explicitly responsible for complying with the terms and conditions of their 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. These regulatory tenets remain foundational to 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 team work to resolve any contentious issues, 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 member's board of directors. If the member continues to be unresponsive, its INPO membership may be suspended. While this option has never been necessary, such action would significantly affect the utility's continued operation, including limiting its ability to obtain insurance.

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 programs.

In return, the industry expects INPO to provide members with results from evaluation, continuous performance monitoring, 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 corrective actions by a member and to verify that the member has implemented the actions.

INPO and its members clearly understand that all parties must maintain the confidentiality of the Institute's reports and related information, and that members must not distribute this information external to their utility. INPO also expects members and participants to use information provided by the Institute to improve nuclear operations and not for other purposes (to gain commercial advantage, for example). Members are to avoid including INPO or INPO documents in litigation.

INPO members that are also members of the collective insurance organization Nuclear Electric Insurance Limited have authorized and instructed the Institute to make available to Nuclear Electric Insurance Limited copies of its evaluation reports and other data. Nuclear Electric Insurance Limited reviews these reports and data for issues that could affect the insurability of its members.

INPO POs&Cs are written with input from, and with the support of, the industry. However, they are written without regard to utility-specific constraints or agreements, such as labor agreements. INPO expects each member to resolve any impediments to the implementation of the POs&Cs that may be imposed by outside organizations.

INPO does not engage in public, media, or legislative activities to promote nuclear power, as such activities could be seen to 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. nuclear industry, including electric utility deregulation and increasing marketplace pressures, the industry has reaffirmed INPO's mission and methods. Even with U.S. utilities now in competition in certain geographical 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 promptly share important information, 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 nearly all of the Institute's programs—plant evaluations, training and accreditation, analysis and information exchange, continuous performance monitoring 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 and 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 Culture

The U.S. nuclear industry believes that a strong safety culture is central to excellence in nuclear plant operations, due to 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 INPO itself, the elements, activities and behaviors that are essential to a strong safety culture are embedded in everything that the Institute 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 entitled, "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 the 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 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 as follows:

- 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

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 pre-evaluation 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, and one-time actions. Several of these are longstanding, 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 POs&Cs, their own experience, and their broad knowledge of industry best practices. This approach shares beneficial industry experience while promoting excellence in the operations, maintenance, and support of operating nuclear plants. Written POs&Cs guide the evaluation process and are the basis for identified areas for improvement. The evaluations focus on those issues that affect nuclear safety and plant reliability.

i. Plant Evaluations

Historically, teams of approximately 18 to 25 qualified and experienced individuals conduct evaluations of operating nuclear plants. In 2018, 34 plant evaluations or WANO peer reviews were conducted by INPO and integrated INPO/WANO-Atlanta Centre teams.

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:

- operational focus
- configuration management
- equipment reliability
- work management
- performance improvement (learning organization)
- organizational effectiveness (leadership, team effectiveness, management)

Teams also evaluate the following foundational areas:

- safety culture
- 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 organizational effectiveness (leadership, teamwork, and management), 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 industry also hosts WANO peer reviews conducted by the WANO-Atlanta Centre. These peer reviews are conducted at each U.S. station approximately every 4 years in place of an INPO plant evaluation. They use the same methodology and performance objectives as that of plant evaluations, but with teams that include international peers.

In June 2018, INPO began a more performance-based approach to evaluation team composition and conduct. The revision to the evaluation approach was based largely on the recognition of improved industry performance, increased confidence in INPO's ability to continuously monitor station performance, and that WANO peer reviews, with a full team complement, remain unchanged and are performed every 4 years.

The team size of INPO performance based evaluations is determined by station performance. A station with exemplary performance may have a base team consisting of six individuals. Stations with lower performance may have teams of 18 to 25 individuals. Base teams are composed of a team leader, organizational effectiveness team leader, INPO exit representative, and three industry peers, including one from the host station.

Guiding principles for the performance based evaluations include the following:

- The scope and composition of evaluations are dictated by performance.
- Operating crews are evaluated in abnormal and emergency conditions.
- Preevaluation observations are conducted during outages or other times when station workload is higher.
- WANO program requirements are fulfilled.
- Team scope and size are adjusted as needed during the evaluation process.
- An overall assessment of station performance is determined.
- The utility CEO is informed of results at an exit meeting.

In 2018, 13 performance-based evaluations, including two international reviews, were conducted. In 2019, of the 31 evaluations and peer reviews scheduled, 16 will be full WANO peer reviews, and 15 will be conducted using the new performance-based approach.

Following the first four performance-based evaluations, a comprehensive self-assessment was performed. While improvements were identified to this new approach, the conclusion was that the performance-based evaluations met the objectives for an INPO evaluation.

The evaluation team continues to provide the utility with formal reports of strengths and areas for improvement, and INPO continues to provide a numerical assessment following each evaluation. Stations are assessed from Category 1 (exemplary) 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 CEOs and managers to know more precisely how their station’s performance compares to the standards of excellence. 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.

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 U.S. industry performance has risen to historically high levels of performance. Numerous improvements have been made in plant safety and reliability by addressing issues identified during evaluations, peer reviews, plant self-assessments and comparison and emulation among plants. The frequency of unplanned shutdowns has decreased markedly, and the reliability and availability of safety systems have improved measurably. The number of stations assessed in the lower categories has substantially declined.

Several U.S. nuclear stations have announced their intention to permanently shut down. As a result, in 2018 INPO began performing shutdown review visits. The objectives of these visits are to determine the readiness of plant personnel to safely shut down the plant and remove the nuclear fuel to its interim storage location. Two of these visits were conducted in 2018, and two are currently scheduled to take place in 2019.

ii. Corporate Evaluations

Member utilities that operate nuclear stations request that INPO conduct corporate evaluations at 6-year intervals. A followup review of corporate performance is conducted 2½ to 3 years following each corporate evaluation to verify progress on identified weaknesses. 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 two corporate evaluations in 2018.

A tailored set of POs&Cs 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 utility's 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 major components, such as steam generator and reactor vessel heads) and station upgrades to extract more power and efficiency

- integrated risk management

- 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. Because of responding to issues identified during corporate evaluations, stations often have refocused appropriate resources and leadership attention on improving station safety and reliability.

INPO provides ongoing oversight and assistance for nuclear corporate organizations between corporate evaluations. Oversight activities include frequent contact with senior corporate executives and corporate visits to observe safety board meetings and other corporate interactions to verify leadership direction, oversight, and engagement in the performance of the members' nuclear stations. Where appropriate to improve corporate performance, assistance is provided, including benchmarking of other high performing member corporate activities.

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 the results of review visits may be used as an input to the evaluation process. The visits are designed as in-depth 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 BWRs and 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 and fuel performance.

Similar to plant evaluations and peer reviews, review visits evaluate station performance against the INPO POs&Cs 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.

Operator Fundamentals Review Visits

In the fall of 2016, INPO identified an adverse trend in operator fundamental events. INPO initiated review visits to target sites that were contributing to the adverse industry performance in operator fundamentals. The purpose of these review visits was to observe operators in training and in-plant settings to determine if weaknesses existed in the execution of operator fundamentals. More than 20 Operator Fundamentals Review Visits were completed in 2017 and 2018. Combined with the industry's implementation of recommendations from an industry operating event report, these review visits have contributed to a reduced number of operator fundamental events and sustained improvement throughout 2018.

Pressurized-Water Reactor 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 EPRI's Steam Generator Management Program, 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-barrier 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 in-depth 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 was expanded to take an even broader look at materials degradation and included flow accelerated corrosion programs and buried pipe and tank integrity.

Boiling-Water Reactor 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 was expanded to take an even broader look at materials degradation and included 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 of the BWR materials review visit.

In 2018, the materials review visit at the corporate office was piloted where the programs were centralized at the utility to look at both BWRs and PWRs. This approach reduces the number of industry peers needed by approximately 18 per year and reduces the number of materials review visit trips from 12 per year to approximately 6 per year.

Alternating Current 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 three to five 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 review visit program with additional focus on program and procedures relied on to prevent, detect, and mitigate LOOP and station blackout events. Team peer selection includes 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, there has been an improving trend in fewer full and partial LOOP events in the industry. The new indicator developed to reflect AC power reliability for the industry and individual sites provides a mechanism to monitor performance. The metric combines LOOP events and emergency diesel generator performance and availability on a 2-year rolling average. Based on improved performance, the AC Power Reliability INPO focus area was transitioned to monitoring status in 2018.

INPO also actively partners with the North American Transmission Forum to develop common expectations and risk assessment tools for the switchyard and grid system interface. In 2014, INPO, the North American Transmission Forum, and EPRI began joint efforts focused on AC power reliability. In 2018, the first pilot review visit was completed that credited the North American Transmission Forum switchyard assist program review of site-specific switchyard programs. 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 per year from the previous average of two per year. These review visits were suspended once industry performance improved and resources were shifted to emergent industry issues.

In 2016, INPO resumed monitoring main generator performance based on an increase in challenges to reliability of generator excitation and stator water cooling systems. Initially, main generator health was reviewed on plant evaluations. Teams focused 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. INPO personnel remain engaged in industry working groups and emergent plant issues related to main generator, turbine, and support systems.

Fuel Integrity Review Visits

INPO used fuel integrity review visits in 2017 and 2018 to gather detailed information regarding fuel integrity performance in the U.S. fleet. Specific sites that experienced a recent fuel failure were chosen, and plans were developed for a site review visit. A team composed of one INPO fuel specialist and two industry peers performed each of the site visits and collected information regarding the causes and corrective actions being taken by each station for its fuel rod failures. Recommendations to improve, and in some cases beneficial practices, are identified and documented in a report that is issued to the station for the results of the visit. Followup on the station response to those recommendations is also performed by INPO personnel.

In mid-2018, INPO issued an industry trend report communicating key causes, corrective action methods, and insights for fuel rod failures based on the results obtained from review visits. Providing this information to stations and utilities enabled all utilities to benefit from the operating experience of others. These review visits and trending by INPO are leading to further action with the nuclear industry to improve fuel integrity performance.

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 ("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 ensure that a systematic process is used to develop and deliver the training. 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
- non-licensed operators
- continuing training for licensed personnel
- shift technical advisors
- instrument and control technicians
- electrical maintenance personnel
- mechanical maintenance personnel
- 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. In addition, a training evaluator was added to the plant evaluation process to provide a more distinct look at the linkage among 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, industrial safety, maintenance, and engineering qualifications. The use of NANTeL increased over the last 2 years with the development of several industry standard courses, resulting in over 3 million individual courses completed over the last 2 years.

Examples of courses and seminars conducted are as follows:

- a nuclear education course for members of boards of 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 shift manager seminar
- first-line leadership seminar
- next-level leadership seminar
- seminars for new department managers

INPO continues to work with the industry to develop and deliver training to address industry needs. For example, INPO recently revised the operations shift manager seminar to incorporate lessons and recommendations identified from industry operating experience.

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.

i. Events Analysis Program

INPO reviews and analyzes operating events from both domestic and international nuclear plants through the Operating Experience Program. The program is designed to provide in-depth 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 INPO Event Report (IER) levels:

- Level 1 IER—Level 1 IERs usually highlight an area of concern important to nuclear safety or plant reliability derived from several significant events but could also be based on a single significant event. Level 1 IER recommendations constitute a new industry standard of performance.
- Level 2 IER—A Level 2 IER highlight an area of concern that may or may not derive from significant events, but have high consequence to plant safety or operation.
- Level 3 IER—A Level 3 IER provides industrywide notification of significant or otherwise important events and associated lessons. Level 3 IERs do not contain recommendations.
- Level 4 IER—A Level 4 IER provides analysis of notable trends of equipment or human performance problems or other industrywide issues.

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 includes safety systems, which have similar components across many stations. A key to this success is the timeliness of reporting. A graded approach is used to prioritize event reports as follows:

- Prompt Reporting—A tentative record is created, shared and sent to INPO for initial screening within 6 INPO business days of the discovery of an event or condition.

- Early Reporting—A tentative record is created, shared and sent to INPO for initial screening within 30 days from discovery of the event.

- Normal Reporting—A complete, final and shared record is created and sent to INPO for screening within 90 days of the event or condition discovery.

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; weaknesses in operations leadership; team effectiveness, and fundamental behaviors; and weaknesses in maintenance technical skills affecting plant operations.

Members should review and take actions, as appropriate, on Level 3 and 4 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 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. These documents are organized in the following categories:

- **Tier One—Excellence Documents** - establish the standards that INPO members and participants are expected to meet. INPO evaluates station performance to the content of these documents. Examples of Tier One documents are as follows:

POs&Cs, which are common for INPO and WANO, are the standards of excellence for plant and corporate performance intended to promote excellence in the operation, maintenance, support and governance of commercial nuclear power plants. The standard is described in the performance or accreditation objective. Supporting criteria provide a breadth and depth to each objective. Member utilities strive to achieve objectives but do not need to meet each specific criterion in an objective to achieve excellence in an area. The POs&Cs 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 describe fundamental attributes, traits and behaviors associated with important industry themes and issues. These principles are generally accepted or time-proven as essential to effective leadership, management and performance of commercial nuclear power plants. In some instances, principles are developed to augment objectives and criteria. Principles may be incorporated into a subsequent revision of objectives and criteria or remain a stand-alone document for a long period.

Level 1 and 2 IERs provide recommendations for actions based on one or more significant industry events, an important industry issue, or an adverse trend. The reports analyze selected events, provide recommendations, and inform the industry of events for inclusion in the operating experience or corrective action program. Stations are expected to implement Level 1 recommendations and, if needed, to develop corrective actions for Level 2 recommendations. Detailed expectations for review and use of IER Level 1 and 2 documents are described in Operating Experience Program Description documents.

- **Tier Two—Supporting and Implementing Documents** - are intended to provide information to assist INPO members and participants in the pursuit of excellence. While it is expected that the intent of these documents be met, strict compliance is not required. Examples of Tier Two documents are as follows:

Level 3 IERs provide prompt notification of important events and reinforcement of related must-know operating experience lessons learned. Station lessons learned should be reviewed for development of internal corrective actions if they are applicable to the reviewing station. Detailed expectations for review and use of Level 3 IER documents are described in Operating Experience Program Description documents. IER Level 3 supplements and appendices are of equivalent pedigree to the parent IER.

Guidelines provide specific information and activities important to achieving standards of excellence as outlined in the related objectives and criteria. The documents provide added levels of guidance and detail considered necessary to implement objectives and criteria but stop short of prescribing specific methods or processes to use.

Process Descriptions reflect the experience gained from operating plants. The information provides a "road map" for how to perform the more advanced, complex and cross-functional activities at stations, which tend to be accomplished by a process. The "AP" annotation originally stood for "advanced plant"; however, the reference has gradually come to refer to "advanced process." These are evolutionary documents that incorporate current best industry practices.

Operating Experience Program Descriptions provide an overview of the INPO-sponsored Operating Experience Program and the program expectations for INPO and INPO members.

- **Tier Three—Other Documents** – include those not addressed in Tier One or Tier Two of this hierarchy. Information in this tier provides reference or amplifying information for various topics to INPO members and participants for their review and discretionary use. The information may be created by an organization other than INPO. Tier Three information varies greatly in format and style and may not be subjected to the strict document production quality controls required for Tier One and Tier Two documents. Examples of Tier Three documents are as follows:

  Level 4 IERs provide analyses of notable trends of equipment or human performance problems or other industrywide issues intended to heighten industry awareness. Stations should use the information in these reports to determine plant vulnerabilities. Detailed expectations for review and use of IER Level 4 documents are described in Operating Experience Program Description documents. IER Level 4 supplements and appendixes are of equivalent pedigree to the parent IER.

  Good Practices provide examples of effective methods for accomplishing elements of nuclear plant management and operation.

  Manuals are collections of data or other information of wide usefulness to INPO members and participants. The documents provide a convenient collection of concepts, insights, and suggested activities of beneficial use to assist station personnel in understanding, implementing and performing a particular station function.

  Reports provide descriptions and results of INPO or INPO-sponsored activities of broad interest to the industry, such as the following:

— information from INPO benchmarking

— information on cumulative analyses of industry events

— information to the industry that does not fall into a specific document type

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 INPO members and participants to meet and exchange information with their counterparts. In 2018, more than 3,400 industry personnel participated in more than 90 seminars, workshops, and technical working meetings 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. Industrywide trends developed from the data are provided to members 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.

Beginning in 1990, the industry has established long-term goals for each indicator on a 5-year interval.

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 industrywide 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 Nuclear Network®. 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.

viii. Plant Performance Indicator

In 2015, INPO created the Plant Performance Indicator, which is a statistical model that provides a numerical value that helps identify a station's current performance and that correlates to a station's assessment score. The Plant Performance Indicator provides an estimate of the current INPO assessment and WANO peer review scores on a quarterly basis. The Plant Performance Indicator is calculated for all plants using input from the Plant Information Center and other external sources. The Plant Performance Indicator also provides sub-models of the most impactful functional and cross-functional areas and provides an estimate of their assessments as well.

While an actual assessment requires a team of evaluators to visit the station under evaluation, the Plant Performance Indicator relies upon station data, rather than direct observation to predict assessment scoring. Because it is based on data that are typically reported quarterly, the Plant Performance Indicator is valuable in that it is generally more current and is updated more frequently than evaluation or peer review team results, which occur approximately every 2 years.

ix. 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 and suggested actions to the industry. Subjects of recent analyses include common contributors to repeat and longstanding equipment problems, adverse trend in primary pump seal failures, adverse trend in debris-related nuclear fuel failures, and weaknesses in handling highly radioactive filters. Stations use this information 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 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 to help the industry achieve a condition in which all stations operate at high levels 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 continuously reviews and analyzes performance data of stations to identify subtle signs of decline. Additionally, a core team of assigned INPO subject matter experts continuously reviews and analyzes performance data pertaining to their specific functional areas. Performance is collaboratively reviewed by all performance monitoring leaders twice a quarter. The INPO senior leadership team reviews and challenges on a quarterly basis the picture of performance presented by the performance monitoring leader. Each performance monitoring leader is responsible for monitoring approximately six stations that are grouped by fleet organizations. When signs of decline are identified, the performance monitoring leader works with station leaders and INPO leadership to develop an assistance plan to arrest the decline and improve performance.

The continuous performance monitoring program has been expanded to include corporate performance. Beginning in 2018, INPO's view of corporate performance is being evaluated and communicated quarterly through a performance summary report. Corporate continuous monitoring has highlighted gaps at several industry corporations that contributed to declines in plant performance. Although early in its implementation, corporate continuous performance monitoring, and its associated INPO and industry assistance, has improved performance at these corporations. Continuous monitoring for the non-U.S. corporations in WANO-Atlanta Centre will begin in 2019.

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

- Monitor: Monitoring leaders use all available data and information to characterize station and corporate performance. Integrating data with plant observations and insights from other touch points allows the performance monitoring leader to develop an integrated picture of station performance. Credible trigger points are used to identify small developing gaps that require attention. Station leaders receive an INPO Performance Summary Report (IPSR) twice each quarter, non-U.S. stations in the WANO-Atlanta Centre receive an updated IPSR every quarter and corporate leaders receive a Corporate IPSR once a quarter. The

IPSR summarizes the current integrated picture of station performance from INPO's perspective.

- 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.

- Intervene: Intervention is required to shape performance improvement using a graded and specific approach. There are three levels of intervention: elevation for narrow shallow gaps; escalation for wider, deeper, or cross functional gaps; and special focus (described in the next sections). 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 additional tools and techniques that rely more on direct observations of station performance and more interactions with station leaders.

e. Member Support Missions

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

INPO personnel and industry peers normally conduct such visits. For example, if a member requests support 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 member support mission visit includes actual methods and plans for improving performance as part of the assistance visit.

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

f. Special Focus Program

There is a direct correlation between station performance and the likelihood of an event, such that very low performing stations typically experience consequential or even significant adverse operational events. To assist these stations in improving performance, INPO created the highest level of member engagement in 2005, termed the Special Focus Program. Since the inception of the program, INPO and WANO-Atlanta Centre have worked together to create methods, tools, and training to sustainably recover performance at these stations. These methods include structured interactions with the site, utility executives, the utility CEO, and the utility Board of Directors. At INPO, a dedicated group of experienced experts is responsible for working with low performing stations directly and engaging the industry to support these stations.

Historically, it was common for 10 to 12 stations to be categorized as low performers and be designated as special focus plants. Some stations would improve performance enough to be removed from the program only to re-enter as performance subsequently degraded. Other stations would linger at low performance levels and remain in the program for several years. As the program methods matured and expertise improved, the number of special focus stations decreased. As of January 2019, there are one U.S. domestic and one WANO-Atlanta Centre international station in the special focus program. Additionally, since 2013, no stations that have emerged from the program have re-entered.

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 operates the WANO-Atlanta Centre, one of the four WANO global regional centers under contract with WANO. The WANO-Atlanta Centre Governing Board appoints an INPO executive to serve as the Atlanta Centre director.

In addition to INPO, WANO-Atlanta Centre members include the following:

- Bruce Power (Canada)
- Centrala Nuclearelectrica (Romania)
- China Huaneng Group (China)
- Comisión Federal de Electricidad (México)
- Emirates Nuclear Energy Corporation (United Arab Emirates)
- Eskom Holdings (South Africa)
- New Brunswick Power (Canada)
- Ontario Power Generation (Canada)
- State Power Investment Corporation (China)

WANO-Atlanta Centre operations and programs are very similar to INPO programs and include the following:

- WANO-Atlanta Centre teams of U.S. and international peers conduct reviews every 4 years at the request of INPO members to identify strengths and areas for improvement associated with nuclear safety and reliability. A WANO-Atlanta Centre 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 Centre, which is then passed to WANO. The operating experience sharing provides detailed descriptions of events and lessons learned to member utilities worldwide. International operating experience sent to WANO is entered into INPO databases and shared with the U.S. members.

- WANO-Atlanta Centre 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 Centre conducts comprehensive continuous monitoring of its international stations in much the same way INPO conducts this function for the U.S. stations.

- WANO-Atlanta Centre conducts member support missions to allow direct sharing of plant operating experience and ideas for improvement.

- WANO-Atlanta Centre, 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 355 units that are operated 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

A coordinated effort of EPRI, INPO, and NEI, in conjunction with senior utility executives, created a joint leadership model to respond 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.

The primary objective of the industry response has been 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 against extreme external events. The U.S. nuclear industry has established 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.

In addition to directly supporting the industry response strategy to the Fukushima accident, INPO issued several 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. Collectively, the IERs contained nearly 100 recommendations, sub-recommendations, and actions. INPO has verified that the actions have been completed through a variety of review activities, including INPO evaluations and WANO peer reviews, Emergency Management Performance Evaluations and in-office document reviews. Sustainability of the actions will be reviewed during WANO peer reviews.

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. In addition, an INPO Good Practice was issued in 2017 to support effective demonstration of diverse and flexible coping strategies.

The INPO emergency plan and emergency response facilities were upgraded in early 2013 and again in 2018 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 NEI and the EPRI, to practice response actions. Some are conducted in conjunction with utilities during their regularly scheduled emergency preparedness drills.

Diverse and Flexible Coping Strategies

NEI and INPO worked with the U.S. nuclear industry to develop a “Diverse and Flexible Coping Strategy,” or FLEX, which 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 on installed equipment, onsite portable equipment, and pre-staged offsite resources. The equipment ranges from diesel-driven pumps and electric generators to ventilation fans, hoses, fittings, cables and communications gear. The new equipment is stored at strategic locations at the sites and protected to ensure that it can be used if other systems that comprise a facility’s multilayered safety strategy are compromised. This flexible approach builds on existing safety systems to protect against unforeseen events.

The concept for offsite support is based on the assumption that onsite resources must be sufficient to cope for the first 24 hours. 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, each site becomes a potential source of FLEX equipment for any other industry site experiencing such an event. During an emergency event, a call to INPO or directly to the other site will activate mobilization of FLEX equipment.

In addition to support from other sites, there are two large response centers—one in Memphis, TN, and one in Phoenix, AZ—that are capable of delivering equipment to any U.S. site within 24 hours. The response centers are managed by a vendor, Strategic Alliance for FLEX Emergency Response. The Pooled Equipment Inventory Company 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 its site-specific response center mobilization manual.

10. Conclusion

The U.S. commercial nuclear industry has made substantial, sustained, and quantifiable improvements in plant safety and performance in the nearly 4 decades since the Three Mile Island accident. 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 with the creation of INPO. 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—*excellence*—in nuclear plant operation.

The U.S. industry commitment to improved performance 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 contributions to the industry it serves.

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 its tradition of sharing and mutual support, conducting itself with utmost integrity and an unrelenting drive to excellent performance.

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APPENDIX B U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: U.S. Nuclear Regulatory Commission, NUREG-1350, "2018–2019 Information Digest," Volume 30, August 2018.

NOTE: Subsequent to the issuance of NUREG-1350, Volume 30, Oyster Creek Nuclear Generating Station and Pilgrim Nuclear Power Station ceased operations, bringing the total to 97 operating nuclear installations in the United States.

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Arkansas Nuclear One 1—Entergy Nuclear Operations, Inc.	PWR	2,568	12/74 05/34
Arkansas Nuclear One 2—Entergy Nuclear Operations, Inc.	PWR	3,026	03/80 07/38
Beaver Valley Power Station, Unit 1—FirstEnergy Nuclear Operating Company	PWR	2,900	10/76 01/36
Beaver Valley Power Station, Unit 2—FirstEnergy Nuclear Operating Company	PWR	2,900	11/87 05/47
Braidwood Station, Unit 1—Exelon Corp., Exelon Generation Corporation, LLC	PWR	3,645	07/88 10/46
Braidwood Station, Unit 2—Exelon Corp., Exelon Generation Corporation, LLC	PWR	3,645	10/88 12/46
Browns Ferry Nuclear Plant, Unit 1—Tennessee Valley Authority	BWR	3,952	08/74 12/33
Browns Ferry Nuclear Plant, Unit 2—Tennessee Valley Authority	BWR	3,952	03/75 06/34
Browns Ferry Nuclear Plant, Unit 3—Tennessee Valley Authority	BWR	3,952	03/77 07/36
Brunswick Steam Electric Plant, Unit 1—Carolina Power & Light Co., Progress Energy	BWR	2,923	03/77 09/36
Brunswick Steam Electric Plant, Unit 2—Carolina Power & Light Co., Progress Energy	BWR	2,923	11/75 12/34
Byron Station, Unit 1—Exelon Corp., Exelon Generation Corporation, LLC	PWR	3,645	09/85 10/44
Byron Station, Unit 2—Exelon Corp., Exelon Generation Corporation, LLC	PWR	3,645	08/87 11/46

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Callaway Plant—AmerenUE, Union Electric Company	PWR	3,565	12/84 10/44
Calvert Cliffs Nuclear Power Plant, Unit 1—Constellation Energy	PWR	2,737	05/75 07/34
Calvert Cliffs Nuclear Power Plant, Unit 2—Constellation Energy	PWR	2,737	04/77 08/36
Catawba Nuclear Station, Unit 1—Duke Energy Carolinas, LLC	PWR	3,411	06/85 12/43
Catawba Nuclear Station, Unit 2—Duke Energy Carolinas, LLC	PWR	3,411	08/86 12/43
Clinton Power Station—Exelon Corporation, Exelon Generation Co., LLC	BWR	3,473	11/87 09/26
Columbia Generating Station—Energy Northwest	BWR	3,544	12/84 12/43
Comanche Peak Nuclear Power Plant, Unit 1—Luminant Generation Company, LLC	PWR	3,612	08/90 02/30
Comanche Peak Nuclear Power Plant, Unit 2—Luminant Generation Company, LLC	PWR	3,612	08/93 02/33
Cooper Nuclear Station—Nebraska Public Power	BWR	2,419	07/74 01/34
Davis-Besse Nuclear Power Station—FirstEnergy Nuclear Operating Co.	PWR	2,817	07/78 04/37
Diablo Canyon Power Plant, Unit 1—Pacific Gas & Electric Co.	PWR	3,411	05/85 11/24
Diablo Canyon Power Plant, Unit 2—Pacific Gas & Electric Co.	PWR	3,411	03/86 08/25
Donald C. Cook Nuclear Plant, Unit 1—Indiana/Michigan Power Co.	PWR	3,304	08/75 10/34
Donald C. Cook Nuclear Plant, Unit 2—Indiana/Michigan Power Co.	PWR	3,468	07/78 12/37
Dresden Nuclear Power Station, Unit 2—Exelon Corporation, Exelon Generation Co., LLC	BWR	2,957	06/70 12/29
Dresden Nuclear Power Station, Unit 3—Exelon Corporation, Exelon Generation Co., LLC	BWR	2,957	11/71 01/31
Duane Arnold Energy Center—FPL Energy Duane Arnold, LLC, Florida Power and Light Co.	BWR	1,912	02/75 02/34

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Edwin I. Hatch Nuclear Plant, Unit 1—Southern Nuclear Operating Co.	BWR	2,804	12/75 08/34
Edwin I. Hatch Nuclear Plant Unit 2—Southern Nuclear Operating Co.	BWR	2,804	09/79 06/38
Fermi, Unit 2—The Detroit Edison Co.	BWR	3,486	01/88 03/45
R.E. Ginna Nuclear Power Plant—Constellation Energy	PWR	1,775	07/70 09/29
Grand Gulf Nuclear Station, Unit 1—Entergy Nuclear Operations, Inc.	BWR	4,408	07/85 11/44
H.B. Robinson Steam Electric Plant, Unit 2—Carolina Power & Light Co.	PWR	2,339	03/71 07/30
Hope Creek Generating Station, Unit 1—PSEG Nuclear, LLC	BWR	3,902	12/86 04/46
Indian Point Nuclear Generating, Unit 2—Entergy Nuclear Operations, Inc.	PWR	3,216	08/74 09/13
Indian Point Nuclear Generating, Unit 3—Entergy Nuclear Operations, Inc.	PWR	3,216	08/76 12/15
James A. FitzPatrick Nuclear Power Plant—Entergy Nuclear Operations, Inc.	BWR	2,536	07/75 10/34
Joseph M. Farley Nuclear Plant, Unit 1—Southern Nuclear Operating Co.	PWR	2,775	12/77 06/37
Joseph M. Farley Nuclear Plant, Unit 2—Southern Nuclear Operating Co.	PWR	2,775	07/81 03/41
La Salle County Station, Unit 1—Exelon Corporation, Exelon Generation Co., LLC	BWR	3,546	01/84 04/42
La Salle County Station, Unit 2—Exelon Corporation, Exelon Generation Co., LLC	BWR	3,546	10/84 12/43
Limerick Generating Station, Unit 1—Exelon Corporation, Exelon Generation Co., LLC	BWR	3,515	02/86 10/44
Limerick Generating Station, Unit 2—Exelon Corporation, Exelon Generation Co., LLC	BWR	3,515	01/90 06/49
McGuire Nuclear Station, Unit 1—Duke Energy Power Company, LLC	PWR	3,411	12/81 06/41
McGuire Nuclear Station, Unit 2—Duke Energy Power Company, LLC	PWR	3,411	03/84 03/43
Millstone Power Station, Unit 2—Dominion Nuclear Connecticut, Inc., Dominion Generation	PWR	2,700	12/75 07/35

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Millstone Power Station, Unit 3—Dominion Nuclear Connecticut, Inc., Dominion Generation	PWR	3,650	04/86 11/45
Monticello Nuclear Generating Plant—Nuclear Management Co.	BWR	2,004	06/71 09/30
Nine Mile Point Nuclear Station, Unit 1—Constellation Energy	BWR	1,850	12/69 08/29
Nine Mile Point Nuclear Station, Unit 2—Constellation Energy	BWR	3,988	03/88 10/46
North Anna Power Station, Unit 1—Virginia Electric & Power Co., Dominion Generation	PWR	2,940	06/78 04/38
North Anna Power Station, Unit 2—Virginia Electric & Power Co., Dominion Generation	PWR	2,940	12/80 08/40
Oconee Nuclear Station, Unit 1—Duke Energy Power Company, LLC	PWR	2,568	07/73 02/33
Oconee Nuclear Station, Unit 2—Duke Energy Power Company, LLC	PWR	2,568	09/74 10/33
Oconee Nuclear Station, Unit 3—Duke Energy Power Company, LLC	PWR	2,568	12/74 12/34
Oyster Creek Nuclear Generating Station—AmerGen Energy Co., LLC, Exelon Corporation	BWR	1,930	12/69 04/29
Palisades Nuclear Plant—Entergy Nuclear Operations, Inc.	PWR	2,565	12/71 03/31
Palo Verde Nuclear Generating Station, Unit 1—Arizona Public Service Company	PWR	3,990	01/86 06/45
Palo Verde Nuclear Generating Station, Unit 2—Arizona Public Service Company	PWR	3,990	09/86 04/46
Palo Verde Nuclear Generating Station, Unit 3—Arizona Public Service Company	PWR	3,990	01/88 11/47
Peach Bottom Atomic Power Station, Unit 2—Exelon Corp., Exelon Generation Corporation, LLC	BWR	4,016	07/74 08/33
Peach Bottom Atomic Power Station, Unit 3—Exelon Corp., Exelon Generation Corporation, LLC	BWR	4,016	12/74 07/34
Perry Nuclear Power Plant, Unit 1—FirstEnergy Nuclear Operating Co.	BWR	3,758	11/87 03/26
Pilgrim Nuclear Power Station, Unit 1—Entergy Nuclear Operations, Inc.	BWR	2,028	12/72 06/32

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Point Beach Nuclear Plant, Unit 1—FLP Energy Point Beach, LLC, Florida Power and Light Co.	PWR	1,800	12/70 10/30
Point Beach Nuclear Plant, Unit 2—FLP Energy Point Beach, LLC, Florida Power and Light Co.	PWR	1,800	10/72 03/33
Prairie Island Nuclear Generating Plant, Unit 1—Nuclear Management Co.	PWR	1,677	12/73 08/33
Prairie Island Nuclear Generating Plant, Unit 2—Nuclear Management Co.	PWR	1,677	12/74 10/34
Quad Cities Nuclear Power Station, Unit 1—Exelon Corporation, Exelon Generation Co., LLC	BWR	2,957	02/73 12/32
Quad Cities Nuclear Power Station, Unit 2—Exelon Corporation, Exelon Generation Co., LLC	BWR	2,957	03/73 12/32
River Bend Station, Unit 1—Entergy Nuclear Operations, Inc.	BWR	3,091	06/86 08/25
Salem Nuclear Generating Station, Unit 1—PSEG Nuclear, LLC	PWR	3,459	06/77 08/36
Salem Nuclear Generating Station, Unit 2—PSEG Nuclear, LLC	PWR	3,459	10/81 04/40
Seabrook Station, Unit 1—FPL Energy Seabrook, LLC	PWR	3,648	08/90 03/30
Sequoyah Nuclear Plant, Unit 1—Tennessee Valley Authority	PWR	3,455	07/81 09/40
Sequoyah Nuclear Plant, Unit 2—Tennessee Valley Authority	PWR	3,455	06/82 09/41
Shearon Harris Nuclear Power Plant, Unit 1—Carolina Power & Light Co.	PWR	2,948	05/87 10/46
South Texas Project, Unit 1—STP Nuclear Operating Co.	PWR	3,853	08/88 08/47
South Texas Project Unit 2—STP Nuclear Operating Co.	PWR	3,853	06/89 12/48
St. Lucie Plant, Unit 1—Florida Power & Light Co.	PWR	3,020	12/76 03/36
St. Lucie Plant, Unit 2—Florida Power & Light Co.	PWR	3,020	08/83 04/43
Surry Power Station, Unit 1—Virginia Electric and Power Co.	PWR	2,587	12/72 05/32

Plant Name and Operating Utility	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Surry Power Station, Unit 2—Virginia Electric and Power Co.	PWR	2,587	05/73 01/33
Susquehanna Steam Electric Station, Unit 1—PPL Susquehanna, LLC	BWR	3,952	06/83 07/42
Susquehanna Steam Electric Station, Unit 2—PPL Susquehanna, LLC	BWR	3,952	02/85 03/44
Three Mile Island Nuclear Station, Unit 1—AmerGen Energy Co., LLC	PWR	2,568	09/74 04/34
Turkey Point Nuclear Generating, Unit 3—Florida Power & Light Co.	PWR	2,644	12/72 07/32
Turkey Point Nuclear Generating, Unit 4—Florida Power & Light Co.	PWR	2,644	09/73 04/33
Virgil C. Summer Nuclear Station—South Carolina Electric & Gas Co.	PWR	2,900	01/84 08/42
Vogtle Electric Generating Plant, Unit 1—Southern Nuclear Operating Co.	PWR	3,625	06/87 01/47
Vogtle Electric Generating Plant, Unit 2—Southern Nuclear Operating Co.	PWR	3,625	05/89 02/49
Waterford Steam Electric Station, Unit 3—Entergy Nuclear Operations, Inc	PWR	3,716	09/85 12/24
Watts Bar Nuclear Plant, Unit 1—Tennessee Valley Authority	PWR	3,459	05/96 11/35
Watts Bar Nuclear Plant, Unit 2—Tennessee Valley Authority	PWR	3,411	10/16 10/55
Wolf Creek Generating Station, Unit 1—Wolf Creek Nuclear Operating Corporation	PWR	3,565	09/85 03/45

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11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission has prepared Revision 7 to NUREG-1650, "The United States of America Eighth National Report for the Convention on Nuclear Safety," for submission for peer review at the eighth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2020. 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 2016, 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.

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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, Transformation, Vienna Declaration, Fukushima.

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