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United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-1650
Revision 8

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

Office of Nuclear Reactor Regulation

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United States Nuclear Regulatory Commission

Protecting People and the Environment

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

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Office of Nuclear Reactor Regulation

ABSTRACT

The U.S. Nuclear Regulatory Commission prepared Revision 8 to NUREG-1650, “The United States of America Ninth National Report for the Convention on Nuclear Safety,” for submission for peer review at the joint eighth and ninth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2023. 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 2019, 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.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

In response to the Coronavirus Disease 2019 (COVID-19) public health emergency, the eighth review meeting, which was scheduled to take place in March 2020, was cancelled. Therefore, the contracting parties agreed to convene a joint eighth and ninth review meeting at the International Atomic Energy Agency in Vienna, Austria, in March 2023.

The U.S. Nuclear Regulatory Commission (NRC) prepared Revision 8 to NUREG-1650, "The United States of America Ninth National Report for the Convention on Nuclear Safety," for submission for peer review at the joint eighth and ninth review meeting of the Convention. 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. The following four U.S. challenges were identified:

(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 eighth U.S. National Report, dated August 2019, including accident tolerant fuel, changes to the Reactor Oversight Process, the backfit process, digital instrumentation and control systems, emergency preparedness rulemaking, risk-informed decisionmaking, subsequent license renewal, and transformation at the NRC. The report also addresses the following safety and regulatory issues that have needed significant attention since 2019:

- advanced reactors
- construction activities at the Vogtle Electric Generating Plant, Units 3 and 4
- data analytics
- licensing, oversight, and facilitation of digital upgrades
- oversight of test reactor fuel event and restart activities
- pandemic response
- risk-informed and performance-based regulations

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

ΔCDP	change in core damage probability
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
CCDP	conditional core damage probability
CEO	chief executive officer
CFR	*Code of Federal Regulations*
CNS	Convention on Nuclear Safety
CNSC	Canadian Nuclear Safety Commission
COVID-19	Coronavirus Disease 2019
CRGR	Committee to Review Generic Requirements
DG	draft regulatory guide
DHS	U.S. Department of Homeland Security
DI&C	digital instrumentation and control
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
IRIS	Industry Reporting and Information System (INPO)
IRRS	Integrated Regulatory Review Service
ISG	interim staff guidance
ITAAC	inspections, tests, analyses, and acceptance criteria
LER	licensee event report
LOCA	loss-of-coolant accident
LR-ISG	license renewal interim staff guidance
MD	management directive
MRP	Materials Reliability Program (EPRI)
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NANTeL	National Academy for Nuclear Training e-Learning
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NEIMA	Nuclear Energy Innovation and Modernization Act
NIMS	National Incident Management System
NIST	National Institute of Standards and Technology
NQA	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
OIG	Office of the Inspector General
OMB	Office of Management and Budget
OSART	Operational Safety Assessment Review Team
POs&Cs	performance objectives and criteria
PPI	Plant Performance Indicator
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RG	regulatory guide
RIDM	risk-informed decisionmaking
RIPE	Risk-Informed Process for Evaluations
RISC	Risk-Informed Steering Committee
RIS	regulatory issue summary
RITSTF	Risk-Informed Technical Specification Task Force
SALTO	Safety Aspects of Long Term Operation
SAMG	severe accident management guideline
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
SLR-ISG	subsequent license renewal interim staff guidance

SRM	staff requirements memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
SSC	structure, system, and component
Sv	sievert
SWP	strategic workforce planning
TI	temporary instruction
TR	topical report
TSTF	Technical Specification Task Force
U.S.	United States
U.S. APWR	U.S. Advanced Pressurized-Water Reactor
VLSSIR	very low safety significance issue resolution process
WANO	World Association of Nuclear Operators
WANO-AC	World Association of Nuclear Operators-Atlanta Centre
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 Ninth 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 joint eighth and ninth 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 eighth U.S. National Report, principally using agency information that is publicly available. This updated report follows the format of the eighth U.S. National Report published in 2019 and is designed to be a standalone document. Therefore, this report duplicates some of the information presented in the 2019 report. To facilitate peer review, Part 1, 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. The summary addresses progress on safety and regulatory issues identified in the 2019 report; progress on outstanding challenges and suggestions; safety and regulatory issues that have arisen since the 2019 report was issued, including strategies used to ensure continued safety of the nuclear installations because of the pandemic; and major accomplishments.

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

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

general safety considerations and summarize major safety-related features. Articles 17 through 19 address the safety of installations.

Similar to the 2019 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 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 (https://www.nrc.gov).

1.2 Changes from the Eighth U.S. National Report

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

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

Report Section		Change
Abstract		Updated to add discussion about the eighth CNS
Executive Summary		Updated to add discussion about the eighth CNS
PART 1		
Section 1	INTRODUCTION	Updated to add discussion about the eighth CNS
1.1	Purpose and Structure of This Report	Updated to add discussion about the eighth CNS
1.2	Changes from the Eighth U.S. National Report	Updated table
Section 2	SUMMARY	Renumbered and updated to add discussion about the eighth CNS
2.1	The U.S. Policy toward Nuclear Activities	Editorial changes only
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	No changes
2.2.1	Reactor Oversight Process	No changes
2.2.2	License Renewal	Updated discussion about units entering the period of extended operation and updated table
2.2.3	Power Uprates	No changes

Report Section		Change
2.2.4	New Reactor Licensing	Updated to clarify licensing process for light-water reactors and all new reactor technologies and updated to provide information on Vogtle Electric Generating Plant, Units 3 and 4; number of applications under review and discussion on international activities
2.3	Safety and Regulatory Issues, and Regulatory Accomplishments	Editorial changes only
2.3.1 *(2.3.1.1–2.3.2.10)*	Safety and Regulatory Issues Discussed in the Eighth U.S. National Report	Updated to discuss current status and activities conducted in the last 3 years
2.3.2 *(2.3.2.1–2.3.2.7)*	Current Safety and Regulatory Issues	Completely updated to address new topics
2.3.3 (2.3.3.1–2.3.3.8)	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 discussion on the 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 ninth CNS
2.4.2	Integrated Regulatory Review Service	Updated to clarify the purpose of the mission
2.4.3	Operational Safety Review Team	Updated to clarify the purpose of the program and include reference to mission results
PART 2		
Article 6	EXISTING NUCLEAR INSTALLATIONS	No changes
6.1	Introduction	Updated to add discussion on most recent NRC budget justification and include reference to performance goals results
6.2	Nuclear Installations in the United States	Updated to include status of plants in operation and shutdown
6.3	Regulatory Processes and Programs	Editorial changes
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	Topic removed. Section renumbered.
6.3.3	Accident Sequence Precursor Program	Section renumbered and updated to discuss issuance of annual report
6.3.4	Operating Experience Program	Section renumbered and updated to discuss the development of the center of expertise
6.3.5	Generic Issues Program	Section renumbered and added reference to office instruction
6.3.6	Rulemaking	Section renumbered and updated to add discussion on opportunities for public participation and openness, procedures for rulemaking plans, and delegated authority.
6.3.7	Fire Protection Regulation Program	Section renumbered and updated to discuss issuance of guidance document and inspection procedure

Report Section		Change
6.3.8	Decommissioning	Section renumbered and updated to discuss decommissioning activities
6.3.9	Reactor Safety Research Program	Section renumbered
6.3.10	Generic Communications and Orders	Section renumbered and updated to discuss recent generic communications issued
6.4	Vienna Declaration on Nuclear Safety	No changes
Article 7	LEGISLATIVE AND REGULATORY FRAMEWORK	No changes
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	No changes
7.2.2	Licensing of Nuclear Installations	Editorial changes only
7.2.3	Inspection and Assessment	No changes
7.2.4	Enforcement	Editorial changes and updated discussion on the maximum civil penalty amount
Article 8	REGULATORY BODY	No changes
8.1	The Regulatory Body	No changes
8.1.1	Mandate	No changes
8.1.2	Authority and Responsibilities	No changes
8.1.2.1	Scope of Authority	Expanded discussion on basis for the NRC's authority
8.1.2.2	The NRC as an Independent Regulatory Agency	Expanded discussion on the NRC's authority
8.1.3	Structure of the Regulatory Body	No changes
8.1.3.1	The Commission	Expanded discussion on the Commission
8.1.3.2	Component Offices of the Commission	Updated 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 offices 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	Updates to more accurately reflect the roles and responsibilities
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 to more accurately describe the NRC's international engagement, and provide a list of missions supported in the last reporting period
8.1.5.1	International Standards	Editorial changes and provided summary of published guidance since the last reporting period
8.1.5.2	Integrated Regulatory Review Service Mission	Updated to clarify the purpose of the mission

Report Section		Change
8.1.5.3	Operational Safety Assessment Review Teams	Updated to clarify the purpose of the mission and 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 2022
8.1.6.2	Human Resources	Updated to include additional 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	Independence of the Regulatory Body and Separation of Functions from those Promoting Nuclear Energy	Changed title and updated throughout
8.3	Ethics Rules Applying to NRC Employees and Former Employees	Editorial changes and updated to include Executive order on ethics
Article 9	RESPONSIBILITY OF THE LICENSE HOLDER	No changes
9.1	Introduction	No changes
9.2	The Licensee's Primary Responsibility for Safety	No changes
9.3	Mechanisms to Enforce Licensee's Responsibilities to Maintain Safety	No changes
9.3.1	Enforcement Program	Editorial changes, updated reference to most recent NRC policy, and updated table of enforcement actions
9.3.2	NRC Petition for Enforcement Process	Editorial changes only
9.3.3	Allegation Program	Updated to provide recent number of allegations and include references
9.4	Openness and Transparency	Updated to discuss licensee notification and reporting requirements
9.5	Financial and Human Resources	No changes
9.5.1	Financial Resources	No changes
9.5.2	Human Resources	No changes
Article 10	PRIORITY TO SAFETY	No changes
10.1	Background	No changes
10.2	Probabilistic Risk Assessment Policy	No changes
10.2.1	Applications of Probabilistic Risk Assessment	Updated to discuss risk-informed initiatives and update references
10.2.2	Level 3 Probabilistic Risk Assessment Project	No changes
10.3	Safety Culture	No changes
10.3.1	Safety Culture Policy Statement	Updated to add reference to IAEA document
10.3.2	NRC Monitoring of Licensee Safety Culture	No changes
10.3.2.1	Background	No changes
10.3.2.2	Enhanced Reactor Oversight Process	Editorial changes and updated references
10.3.3	NRC Safety Culture	Updated references
10.4	Managing the Safety and Security Interface	Editorial changes only
Article 11	FINANCIAL AND HUMAN RESOURCES	No changes
11.1	Financial Resources	Editorial changes only

Report Section		Change
11.1.1	Financial Qualifications for Construction and Operations	Updated to add discussion on rulemaking activities
11.1.1.1	Construction Permit Reviews	No changes
11.1.1.2	Operating License Reviews	No changes
11.1.1.3	Combined License Application Reviews	No changes
11.1.1.4	Reviews of License Transfers	No changes
11.1.2	Financial Assurance for Decommissioning	Updated to remove discussion on rulemaking activities.
11.1.3	Financial Protection Program for Liability Claims Arising from Nuclear Incidents	Updated to indicate next adjustment of financial protection, indicate insured amounts, and reference to the most recent Price-Anderson report
11.1.4	Insurance Program for Onsite Property Damages Arising from Nuclear Incidents	Updated to include amount for property insurance
11.2	Regulatory Requirements for Qualifying, Training, and Retraining Personnel	Updated to include more description on operators' training, requirements and guidance
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	No changes
12.1	Overview of Regulatory Requirements	No changes
12.2	Regulatory Review and Control Activities	No changes
12.2.1	Nuclear Power Plant Design and Modifications and Operator Actions	Updated references and listed modifications approved for power increases
12.2.2	Organizational Issues	Updated to reflect training programs in the last reporting cycle
12.2.3	Emergency Operating Procedures and Plant Procedures	Updated status on post-Fukushima activities and updated information on mitigating strategies rule
12.2.4	Shift Staffing	Added discussion on staffing for 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	Updated to describe recent activities
12.2.7	Fitness for Duty	Updated to describe the impact of COVID-19 on exemptions
12.3	Licensee Human Factors Program	No changes
12.4	Feedback and Experience	No changes
12.4.1	Human Factors Associated with Digital Instrumentation and Control	Updated throughout
12.4.2	Human Performance in Decommissioning Activities	Updated reference to rulemaking
12.4.3	Human Performance Research	No changes
Article 13	QUALITY ASSURANCE	No changes
13.1	Background	No changes
13.2	Regulatory Policy and Requirements	No changes
13.2.1	Appendix A to 10 CFR Part 50	No changes
13.2.2	Appendix B to 10 CFR Part 50	No changes
13.2.3	Approaches for Adopting More Widely Accepted International Quality Standards	No changes
13.3	Quality Assurance Regulatory Guidance	No changes

Report Section		Change
13.3.1	Guidance for Staff Reviews for Licensing	No changes
13.3.2	Guidance for Design and Construction Activities	No changes
13.3.3	Guidance for Operational Activities	No changes
13.4	Quality Assurance Programs	No changes
13.5	Quality Assurance Audits Performed by Licensees	No changes
13.5.1	Audits of Vendors and Suppliers	No changes
13.6	Vendor Inspection Program	Updated references
Article 14	ASSESSMENT AND VERIFICATION OF SAFETY	No changes
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	Editorial changes and updated references
14.1.3	Power Uprates	No changes
14.1.3.1	Governing Documents and Process	Editorial changes only
14.1.3.2	Experience	Updated discussion on power uprates approved in the last reporting cycle
14.1.4	License Renewal	No changes
14.1.4.1	Governing Rules, Documents, and Process	Revised title and 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 discussion on new guidance
14.1.5	The United States and Periodic Safety Reviews	Editorial changes and updated references
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, Forward Fitting, and Issue Finality Processes: Timely Imposition of New Requirements	Changed title and updated to clarify the process and scope, and clarified role of the Committee to Review Generic Requirements.
14.1.5.3	License Renewal Confirms Safety of Plants	No changes
14.1.5.4	Risk-Informed Regulation and the Reactor Oversight Process	Updated to include references
14.1.5.5	Licensee Responsibilities for Safety: Regulations and Initiatives Beyond Regulations	No changes
14.1.5.6	The NRC's Regulatory Process Compared with International Safety Reviews	Updated information on peer review missions and other international participations
14.2	Verification by Analysis, Surveillance, Testing, and Inspection	Updated to clarify updates to 10 CFR 50.55a
14.3	Vienna Declaration on Nuclear Safety	No changes
Article 15	RADIATION PROTECTION	No changes
15.1	Overview of Regulatory Requirements and Authority	No changes

Report Section		Change
15.2	Regulatory Framework and Expectations	Added discussion on 10 CFR Part 37, modified description on 10 CFR Part 71, and made editorial changes
15.3	Radiation Protection Activities and Control of Radiation Exposure	Editorial changes only
15.3.1	Control of Radiation Exposure of Occupational Workers	Updated collective doses
15.3.2	Control of Radiation Exposure of Members of the Public	No changes
Article 16	EMERGENCY PREPAREDNESS	No changes
16.1	Emergency Plans and Programs	No changes
16.1.1	Background and Overview of Regulatory Requirements	Updated to clarify guidance for public protection
16.1.2	National Response to an Emergency	Updated references
16.1.2.1	Federal Response	Editorial changes only
16.1.2.2	Licensee, State, Tribal, and Local Response	No changes
16.1.2.3	NRC Response	Updated to clarify incident response program
16.1.2.4	Aspects of Security that Support Response	No changes
16.1.3	Implementation of Emergency Preparedness Measures	No changes
16.1.3.1	Emergency Classification System and Emergency Action Levels	Updated references and summarized severe accidents discussion
16.1.3.2	Offsite Emergency Planning and Preparedness	Editorial changes only
16.1.3.3	Emergency Preparedness Facilities	Editorial change and added references
16.1.3.4	Recommendations for Protective Action in Severe Accidents	No changes
16.1.4	Emergency Response Exercises	Updated to include frequent exercise performed
16.1.5	Regulatory Review and Inspection Practices	Updated to clarify the oversight process and removed discussion on FEMA
16.2	Communications Activities	No change
16.2.1	Communications with Neighboring States and International Arrangements	Updated information on agreements and added discussion on observation of exercises
16.2.2	Communications with the Public	No changes
Article 17	SITING	Updated to summarize post-Fukushima impact
17.1	Background	No changes
17.2	Safety Elements of Siting	No changes
17.2.1	Background	Updated status of guidance and updated references
17.2.2	Assessments of Non-seismic Aspects of Siting	Updated references and described guidance
17.2.3	Assessments of Seismic and Geological Aspects of Siting	Updated and added new references
17.2.4	Assessments of Radiological Consequences from Postulated Accidents	Updated references and uses of guidance for license reviews
17.3	Environmental Protection Elements of Siting	No changes
17.3.1	Governing Documents and Process	Editorial changes and updates to references
17.3.2	Other Considerations for Environmental Reviews	Editorial changes and added references

Report Section		Change
17.4	Reevaluation of Site-Related Factors	Updated description on seismic requirements and added references
17.5	Consultation with Other Contracting Parties To Be Affected by the Installation	No changes
17.6	Vienna Declaration on Nuclear Safety	No changes
Article 18	DESIGN AND CONSTRUCTION	No changes
18.1	Implementation of Defense-in-Depth	No changes
18.1.1	Overview of Regulatory Requirements and Governing Documents	Updated references
18.1.2	Application of the Defense in Depth Philosophy	Editorial changes only
18.1.3	Regulatory Review and Control Activities	Updated throughout
18.1.4	Experience and Implementation of Defense-in-Depth Measures	Updated discussion on implementation of lessons from the Fukushima accident
18.2	Technologies Proven by Experience or Qualified by Testing or Analysis	Updated discussion on small modular reactors
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 and updated references
18.3.2.1	Human Factors Engineering	No changes
18.3.2.2	Digital Instrumentation and Controls	Updated to reflect current experience, lessons learned, and added guidance. Updated international participation.
18.3.2.3	Cybersecurity	Updated to reflect current experience and references, included description of research project and relations with other agencies
18.4	New Reactor Construction Experience Program	Editorial changes only
18.5	Vienna Declaration on Nuclear Safety	No changes
Article 19	OPERATION	No changes
19.1	Initial Authorization to Operate	Editorial changes and included applications approval status
19.2	Definition and Revision of Operational Limits and Conditions	Updated discussion on technical specifications and updated references
19.3	Approved Procedures	Updated to add requirement on quality assurance process
19.4	Procedures for Responding to Anticipated Operational Occurrences and Accidents	No changes
19.5	Availability of Engineering and Technical Support	Editorial changes only
19.6	Incident Reporting	Updated references and discussion on significant reactor events
19.7	Programs To Collect and Analyze Operating Experience	Updated references and discussion on data collection and operating experience
19.8	Radioactive Waste	Updated references, waste amounts, and repository discussion
19.9	Vienna Declaration on Nuclear Safety	No changes

Report Section	Change
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	Updated references
APPENDIX B—U.S. COMMERCIAL NUCLEAR POWER REACTORS	Updated to reflect plant shutdowns

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 7, "The United States of America Eighth National Report for the Convention on Nuclear Safety," dated August 2019). 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 NRC as an independent agency of the Federal Government. The agency's mission is to license and regulate the Nation's civilian use of radioactive materials to provide reasonable assurance of 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, including nonpower production and utilization facilities, 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 https://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 relevant national policy regarding such activities is transparent and available to everyone.

2.1.2 Regulatory Body Challenges

The NRC identified major challenges for the future in NUREG-1614, "Strategic Plan: Fiscal Years 2022-2026," Volume 8, dated September 2021. Many external factors, including the following, influence the ability of the NRC to achieve its strategic goals and the associated strategic objectives:

- market forces and climate change mitigation
- globalization and development of nuclear technology
- security threats and significant incidents
- government and regulatory impacts
- international treaties and conventions
- workforce dynamics
- information technology advances

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 of this report) identifies the agency's most serious management and performance challenges. OIG-22-A-01, "Inspector General's Assessment of the Most Serious Management and Performance Challenges Facing the Nuclear Regulatory Commission in Fiscal Year 2022," dated October 12, 2021, 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) 2022 management and performance challenges are the following:

- ensuring safety while transforming into a modern, risk-informed regulator

- regulatory oversight of the decommissioning process and managing decommissioning trust funds

- using Coronavirus Disease 2019 (COVID-19) lessons learned to strengthen NRC readiness to respond to future mission-affecting disruptions

- readiness to license and regulate new technologies and maintaining the integrity of the associated intellectual property

- ensuring the safe and effective acquisition, management, and protection of information technology and data

- strategic workforce planning during transformation and industry change

- oversight of materials, waste, and the National Materials Program

- management and transparency of financial and acquisitions operations

- readiness to address cyberthreats to critical national infrastructure sectors impacting the NRC's public health and safety mission or NRC licensees

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. Each cornerstone contains performance indicators to ensure that their objectives are being met. Satisfactory licensee performance in the cornerstones provides reasonable assurance of safe facility operation and that the NRC's safety mission is being accomplished.

Inspection reports, including the results of emergency exercise evaluations, are on the NRC public Web site at https://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 2022, 84 of the 92 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 of August 2022, nine additional units have entered the period of extended operation, as seen below. Although 10 units with renewed licenses have shut down, a total of 52 units are currently operating beyond 40 years.

Table 2 Units that Entered the Period of Extended Operation

Year 2020	Year 2021	Year 2022
• Salem Nuclear Generating Station, Unit 2 • North Anna Power Station, Unit 2 • Sequoyah Nuclear Plant, Unit 1	• Joseph M. Farley Nuclear Plant, Unit 2 • McGuire Nuclear Station, Unit 1 • Sequoyah Nuclear Plant, Unit 2	• La Salle County Station, Unit 1 • Susquehanna Steam Electric Station, Unit 1 • Virgil C. Summer Nuclear Station

Section 2.3.1.9 of this report discusses subsequent license renewal (i.e., renewal up to 80 years). Article 14 of this report discusses the license renewal process in detail, including the pending 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. The NRC focuses on safety as part of the review for power uprates. 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, preapplication and readiness reviews for current and future new reactor licensing, and infrastructure development to support oversight and licensing for all new reactor technologies. The NRC is in the process of completing ongoing licensing reviews; overseeing construction activities associated with two new reactor units (Vogtle Electric Generating Plant, Units 3 and 4) 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 establishing the regulatory framework for 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 and awareness for new reactor designs, strengthen reactor siting reviews, and improve the effectiveness and efficiency of inspections that continue to enhance the agency's ability to collect and share construction experience.

The NRC staff is interacting with vendors and utilities on new reactor applications and licensing activities. The NRC staff is actively reviewing topical reports associated with potential design certification and construction permit applications. While some licensing activities use the licensing process specified in 10 CFR Part 50, "Domestic Licensing of Production and Utilizations Facilities," most current activities under review follow the licensing process specified in 10 CFR Part 52. This 10 CFR Part 52 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 the Multinational Design Evaluation Programme, a unique international forum that includes representatives from the regulatory authorities of Argentina, Canada, China, Finland, France, Hungary, India, Japan, the Republic of Korea, the Russian Federation, South Africa, Turkey, 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 Programme.

The Multinational Design Evaluation Programme interacts with various representatives from the industry, including vendors and operators, standards development organizations, the Western European Nuclear Regulators' Association, the International Atomic Energy Agency (IAEA), and the World Nuclear Association. The activities of the Multinational Design Evaluation Programme in which the NRC participated 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) activities to harmonize and converge on regulatory practices in the area of vendor inspection cooperation.

At the end of 2021, the Multinational Design Evaluation Programme sunset several design-specific working groups, including the AP1000 and the APR1400 groups, because these reactors have completed 2 years of initial operations. In addition, the vendor inspection cooperation activities were transferred to a broader forum of cooperation under the NEA. Therefore, 2021 marked the end of the NRC's membership in the program.

The NRC chairs the Small Modular Reactor Regulators' Forum, which is an international forum that enhances nuclear safety by identifying and resolving common safety issues that may challenge regulatory reviews associated with these new reactors. The forum includes representative members from the regulatory authorities from Canada, China, Finland, France, the Republic of Korea, Russian Federation, Saudi Arabia, South Africa, Japan, and the United Kingdom. The NRC participates in the NEA's Committee on Nuclear Regulatory Activities, which has various working groups focused on licensing new and advanced reactors and is cooperating with the IAEA on its assessment of the applicability of current safety standards to advanced reactors and novel technologies.

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 eighth 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 Eighth U.S. National Report

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

2.3.1.1 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 enhance the tolerance under severe beyond-design-basis accidents; permit higher burnup and enrichment; and improve performance and related economics under normal operations. 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 substitute claddings intended to reduce corrosion, increase wear resistance, and reduce the production of hydrogen under accident conditions. To offset the added cost of accident tolerant technologies, the U.S. nuclear industry is also pursuing increases in burnup and enrichment levels beyond those that have been currently approved, which could allow for longer operating cycles between refueling outages.

To support licensing activities, 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," which describes a new paradigm for fuel licensing. Version 1.2 of the plan, dated September 2021, reflects the industry's increased focus on licensing higher burnup and increased enrichment fuels. 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, higher burnup, and increased enrichment fuel designs. The staff believes that adherence to this strategy, which encourages significant engagement 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, higher burnup, and increased enrichment fuel designs.

A summary of the NRC activities to prepare for the licensing of accident tolerant fuels is available on the NRC public Web site at https://www.nrc.gov/reactors/atf.html.

2.3.1.2 Assessment of Debris Accumulation on Sump Performance

Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," 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. On July 23, 2019, by memorandum entitled "Closure of Generic Issue GI-191, 'Assessment of Debris Accumulation on PWR Sump Performance,'" the staff closed GSI-191 because the technical issues identified were well understood and most licensees had addressed all safety-significant issues. Even though GSI-191 is closed, each plant must still respond to GL 2004-02.

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 generally 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. The staff developed “U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses,” dated September 4, 2019, to guide its review of licensee submittals of in-vessel issues. This guidance describes a risk-informed method for evaluating the issue commensurate with its safety significance for each plant configuration.

For boiling-water reactors (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 knowledge of the phenomena in PWR strainer and downstream blockage issues was updated. 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.

2.3.1.3 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.

The staff recently implemented the very low safety significance issue resolution process (VLSSIR) to improve existing NRC processes so that certain very low safety significance issues that involve licensing-basis questions are promptly resolved without an excessive use of resources, thereby enabling the NRC and licensees to better focus resources on issues of greater safety significance. The process has been used several times, and an effectiveness review concluded that the process is working as intended.

The NRC has worked with stakeholders to develop streamlined processes and procedures for inspecting engineering programs. The NRC staff proposed modifying inspection procedures and changing the engineering inspection cycle from triennial to quadrennial. This proposal is described in SECY-18-0113, "Recommendations for Modifying the Reactor Oversight Process Engineering Inspections," dated November 13, 2018. In August 2021, the Commission approved the withdrawal of SECY-18-0113 to allow the staff to consider new information and feedback from internal and external stakeholders, including inspectors, members of the public, and the nuclear industry. On June 7, 2022, based on additional review, the NRC staff submitted SECY-22-0053, "Recommendation for Modifying the Periodicity of Reactor Oversight Process Engineering Inspections," to the Commission requesting approval to revise the engineering inspection program. The staff also continues to assess and improve the Reactor Oversight Process as part of its transformation activities (discussed in Section 2.3.10 of this report), using stakeholder correspondence, and feedback from public meetings and the Reactor Oversight Process Self-Assessment Program. Significant changes to the Reactor Oversight Process will be submitted to the Commission for either approval or notification in accordance with Management Directive 8.13, "Reactor Oversight Process," before the end of FY2022.

The NRC staff completed a "Comprehensive Review of the Reactor Oversight Process Problem Identification and Resolution Inspection Program," dated November 12, 2020. Recommended revisions to this inspection program seek to improve inspections to verify that licensees are identifying, evaluating, and correcting issues. These recommendations are being considered for implementation at the beginning of the 2024 Reactor Oversight Process biennial cycle. The staff also completed an effectiveness review of the cross-cutting issues process, recommending minor changes to the process to make it more proactive in identifying cross-cutting concerns before they lead to more safety significant issues. The staff discusses the recommendations in a memorandum entitled "Dispositioning of Cross-Cutting Issues Program Effectiveness Review Recommendations," dated September 17, 2021. Implementation of those recommendations that were approved is in progress.

2.3.1.4 Clarifying the Backfitting, Forward Fitting, and Issue Finality Processes

Backfitting for nuclear power reactors licensed under 10 CFR Part 50 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 requirements or the imposition of a regulatory staff position interpreting the Commission's regulations or requirements 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 licensed under 10 CFR Part 50 are found in 10 CFR 50.109, "Backfitting." Backfitting requirements for materials licensees are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"; 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"; and 10 CFR Part 76, "Certification of Gaseous Diffusion Plants."

The NRC's "issue finality" requirements in 10 CFR Part 52 are intended to accomplish the same objective as those in 10 CFR 50.109 by providing criteria that the NRC or an applicant must satisfy to change an early site permit, standard design certification, combined license, standard design approval, or manufacturing license. If the NRC, or an applicant referencing a 10 CFR Part 52 approval in its application, proposes to change an existing 10 CFR Part 52 approval, then the NRC or applicant must follow a clearly defined process. Issue finality provides a degree of stability to these approvals just as backfitting provides regulatory stability in 10 CFR Parts 50, 70, 72, and 76. It also provides greater certainty and efficiency in the licensing process for those applicants choosing to incorporate by reference a 10 CFR Part 52 approval.

The NRC is updating 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 based on 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 changes.

The changes were principally aimed at implementing a more rigorous and disciplined backfitting process. In the 2019 U.S. National Report, the NRC reported significant progress in this effort, including the conduct of backfitting trainings and workshops. On September 20, 2019, the NRC issued Management Directive (MD) 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests." This document contains the Commission's policies on backfitting, issue finality, and forward fitting. Forward fitting occurs when the NRC conditions its approval of a licensee-initiated request for a licensing action on the licensee's compliance with a new or modified requirement or staff interpretation of a requirement that the licensee did not request. Generally, the new or modified requirement or staff interpretation must result in a change to the licensee's systems, structures, components, design, procedures, or organization. A similar process for forward fitting can also apply to certain applications for initial licenses. As with backfitting, the NRC must justify forward fitting actions. Unlike backfitting, the forward fitting requirements for the NRC are not in the NRC's regulations but appear in MD 8.4.

Based on the Commission's backfitting, issue finality, and forward fitting policies in MD 8.4, the NRC staff revised its guidance. On March 23, 2020, the NRC issued NUREG-1409, Revision 1, "Backfitting Guidelines: Draft Report for Comment," the first update to NUREG-1409 since its issuance in 1990. The NRC received 13 comment submittals (with approximately 250 individual comments) from the public. Based on the comments, the NRC staff made significant changes to draft NUREG-1409, Revision 1, and provided it to the Commission for its consideration.

The NRC also created a backfitting, issue finality, and forward fitting community of practice to promote the sharing and consistent implementation of backfitting, forward fitting, and issue finality issues, knowledge, and practices across the agency. The community of practice includes backfitting subject matter experts in every region and most program offices. The NRC also created an internal SharePoint site to share backfitting knowledge, experience, and lessons learned.

Section 14.1.5.2 of this report provides additional information about the NRC's backfitting, forward fitting, and issue finality processes.

2.3.1.5 Digital Instrumentation and Control Systems

The staff continues to make progress in 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.

Examples of accomplishments in this area include development of (1) guidance for applicants and licensees implementing digital instrumentation and control changes without prior NRC approval, (2) staff guidance for making licensing decisions on digital upgrades to operating reactors earlier in the licensee's life-cycle software development process, (3) staff guidance for a graded approach to address diversity and defense in depth against potential common-cause failure, and (4) staff guidance for technology-neutral, nonlight-water reactor digital designs. Section 2.3.2.4 of this report provides additional information about the regulatory enhancements to facilitate analog or digital upgrades. The NRC is implementing the licensing, certification, and oversight of several major digital instrumentation and control projects planned by the U.S. nuclear industry under the enhanced and modernized infrastructure.

In parallel, the NRC staff also licensed new digital instrumentation and control systems for operating plants and evaluated several new reactor applications that fully incorporate highly integrated digital technologies. Examples include (1) approval of the NuScale small modular reactor instrumentation and control system, (2) certification approval of the APR1400 instrumentation and control system, (3) approval of a core protection calculator upgrade for Waterford Steam Electric Station, Unit 3, and (4) generic approval of several digital instrumentation and control platforms.

Sections 2.3.3.5 and 18.3.2.2 of this report provide additional details on these efforts.

2.3.1.6 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,[2] 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 Information Notice (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.[3] 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 design features.

In response to this operating experience, some licensees implemented plant modifications through amendments to their operating licenses. Other licensees, except South Texas Project Nuclear Operating Company,[4] have chosen to implement the Nuclear Energy Institute's (NEI's) voluntary industry initiative, which is discussed in NEI letters dated October 9, 2013; March 16, 2015; September 20, 2018; and June 6, 2019. 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 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. For plants that are susceptible to open phase conditions, the licensees have completed plant modifications. In March 2017, the Commission directed the NRC staff to "verify that licensees have appropriately implemented the voluntary industry initiative" and that each licensee has "satisfactory implementation of the technical resolution."

To evaluate the adequacy of the open phase isolation system designs, the NRC staff issued Temporary Instruction (TI) 2515/194, "Inspection of the Licensees' Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerabilities in Electric Power

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

3 South Texas Project, Unit 2, 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.

4 South Texas Project's analysis demonstrated, and the NRC staff verified, that the plant's electrical power system is capable of mitigating an open phase in a timely manner.

Systems (NRC Bulletin 2012-01)," dated October 31, 2017. In 2018, the NRC staff used TI 2515/194 to conduct pilot inspections at four nuclear power plants representing four fundamental open phase isolation system designs. Subsequently, NRC regional inspectors used TI 2515/194 to verify the implementation of the voluntary industry initiative at other nuclear power plants.

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.

In June 2019, the NEI submitted a revision to the voluntary industry initiative, which referenced NEI 19-02, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights," Revision 0, dated May 2019. The revised voluntary industry initiative includes an option for not enabling the automatic functions of the open phase isolation system to isolate a power supply affected by an open phase condition based on assessing the change in risk between operating with automatic functions versus reliance on operator manual action. The NRC staff conducted audits at two pilot sites to assess the implementation of the revised voluntary industry initiative.

In August 2020, the NRC revised TI 2515/194 to include instructions for verifying that licensees appropriately implemented the revised voluntary industry initiative with risk analyses in accordance with the guidance in NEI 19-02. The NRC staff continues to perform inspections using TI 2515/194 to verify that the licensees implemented the voluntary industry initiative to address the open phase vulnerabilities. Once the inspection at a site using TI 2515/194 is completed satisfactorily, the NRC staff sends a letter to the licensee closing NRC Bulletin 2012-01 for that site. The staff is making good progress with this action as it has issued bulletin closure letters for approximately 95 percent of the operating plants. The staff anticipates that NRC Bulletin 2012-01 will be closed for all operating plants by the fourth quarter of calendar year 2022.

In July 2021, the NRC revised several Reactor Oversight Process inspection procedures and Inspection Manual Chapters to provide periodic oversight of licensee's implementation of the voluntary industry initiative to address the open phase vulnerabilities.

The NRC provides additional information on open phase conditions on the NRC's public Web site at https://www.nrc.gov/reactors/operating/ops-experience/open-phase-electric-systems.html.

2.3.1.7 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, nonlight-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 examined these issues and recommended a rulemaking to revise the regulations.

This rulemaking would amend the NRC's regulations to add alternative emergency preparedness requirements for small modular reactors and other new technologies. The final rule would be technology inclusive and would provide existing and future light-water small modular reactors, nonlight-water reactor applicants and licensees, certain existing nonpower production and utilization facilities, and nonpower production and utilization facilities licensed after the effective date of the final rule, with the alternative to develop a performance-based emergency preparedness program rather than using the existing deterministic emergency preparedness requirements in 10 CFR Part 50.

On May 12, 2020, the NRC published in Volume 85 of the *Federal Register* (FR), page 28436 (85 FR 28436), the proposed rule, "Emergency Preparedness for Small Modular Reactors and Other New Technologies," for a 75-day public comment period. On May 29, 2020, the NRC published a notice to correct the definition of "non-power production or utilization facility" (85 FR 32308).

The NRC received several requests to extend the comment period by 6 months or more because of the COVID-19 public health emergency. On July 21, 2020, the NRC extended the comment period by 60 days, and the public comment period ended on September 25, 2020 (85 FR 44025). The NRC received comments from 2,212 individuals and organizations, including 2,087 form letters. The staff's analysis identified 649 unique comments on the proposed rule and associated guidance, the regulatory analysis, and the environmental assessment. The commenters included State and local governments, Tribal governments and Tribal organizations, Federal agencies, members of the nuclear power industry, nongovernmental organizations, and private citizens. The NRC staff used these comments to develop the draft final rule.

On January 3, 2022, the draft final rule package was submitted to the Commission for its consideration as SECY-22-0001, "Final Rule: Emergency Preparedness for Small Modular reactors and Other New Technologies." The draft final rule includes the following provisions:

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

- a requirement for a hazard analysis of any NRC licensed or nonlicensed facility contiguous to or near a small modular reactor or other new technologies, that considers any hazard that would adversely impact the implementation of emergency plans developed under this framework

- 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 offsite capabilities and resources available to prevent contaminated food and water from entering the ingestion pathway

The new NRC emergency preparedness requirements in the draft final rule and implementing guidance adopt a consequence-oriented, risk-informed, performance-based, and technology-inclusive approach.

If approved by the Commission, the final rule will be effective 30 days after publication in the *Federal Register*.

2.3.1.8 Risk-Informed Decisionmaking

The NRC is advancing the use of quantitative and qualitative risk information in its decisionmaking processes while focusing on the five key principles of RIDM, which include the need to be consistent with the defense-in-depth philosophy, maintain sufficient safety margins, and ensure appropriate performance monitoring. Activities, such as those discussed below, include an assessment of challenges to further progress in RIDM and the implementation of communication strategies and guidance development efforts. These initiatives will reinforce the expectation to use risk insights at the early stages of regulatory activities to more efficiently guide the agency's efforts, improve communications, and achieve consistency. The NRC's vision is that RIDM—and notably, the resulting safety focus and efficiency benefit—will be applied broadly across regulatory activities beyond the nuclear reactor safety program.

<u>NRC's Risk-Informed Steering Committee</u>. The Risk-Informed Steering Committee (RISC) was an NRC senior management committee that provided 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" (60 FR 42622; August 16, 1995). The NRC suspended its RISC meetings in 2019 because (1) the RISC had evolved to a point that it was primarily focused on addressing non-strategic items such as reactor-specific licensing issues involving probabilistic risk assessment (PRA) tools and methodology, and (2) other agency-level initiatives (i.e., Futures Assessment and Be riskSMART), some of which are further discussed in Sections 2.3.1.10 and 2.3.2.7 of this report, had superseded topics previously under the purview of the RISC.

The nuclear power industry has its own RISC, which is a counterpart to the NRC's committee. Since the suspension of the NRC RISC, the NRC RISC Chair and subject matter experts have continued to interact with the industry's RISC on a recurring basis to discuss RIDM issues. The NRC is continuing to explore ways to enhance these interactions to support meaningful change and results as they relate to advancing RIDM in the agency and industry.

<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 reviewing requested 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 quantitative metrics to one that considers qualitative risk insights in decisionmaking along with defense-in-depth philosophy, safety margins, performance measurement strategies, and regulatory compliance. Recent key accomplishments in this area include knowledge transfer activities that were performed at the staff level on the expanded use of RIDM and integrated review teams for licensing actions, and a risk forum that involved an exchange of RIDM knowledge between the NRC and industry participants. Another key accomplishment 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.

<u>Improving Use of Risk Information in Licensing Actions</u>. The RIDM action plan recognizes that improvements to licensing processes should specify how risk information is used in reviewing licensing action requests. The NRC plans to enhance the existing framework in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," to use a graded approach.

The NRC has sought to optimize the review process for formal risk-informed licensing actions, submitted in accordance with RG 1.174, Revision 3, dated January 2018. Additionally, the NRC updated guidance in Office of Nuclear Reactor Regulation Office Instruction LIC-206, "Integrated Risk-Informed Decision-Making for Licensing Reviews," Revision 1, dated June 26, 2020 for licensing actions that do not strictly adhere to RG 1.174, Revision 3. The objective of the updated guidance is to more effectively support the NRC staff in considering risk insights in licensing reviews through the establishment of integrated review teams, where risk analysts work together with traditional deterministic reviewers to complete these technical evaluations. Integrated review teams are better able to access tools to use risk information to tailor the focus, depth, and scope of reviews. The updated guidance also provides RIDM implementation guidance to technical reviewers and other useful resources.

U.S. nuclear utilities are actively pursuing efforts to adopt RIDM tools within their licensing basis to gain operational and engineering flexibilities. For example, in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," the NRC provides requirements for implementing a process to risk-inform the characterization of SSCs. This 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. 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. The NRC staff has approved several licensing applications requesting adoption of 10 CFR 50.69.

The NRC has also reviewed licensing applications involving the establishment of a risk management approach for certain surveillance frequencies and 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. These initiatives are intended to maintain and improve safety by incorporating risk assessment and management techniques in the technical specifications while reducing unnecessary burden. Licensees continue to submit licensing applications requesting adoption of TSTF-505 and TSTF-425, and the NRC staff has approved several applications.

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.

In April 2017, NRC staff completed the action to consider these technical issues and it reported that it is well-positioned to review subsequent license renewal applications. Currently, the staff examines these issues on a case-by-case basis for each subsequent license renewal application.

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.” To support 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. Specifically, in July 2017, the NRC issued 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,” which provide guidance to NRC staff reviewing a subsequent license renewal application to ensure that it meets the requirements of 10 CFR Part 54. There was significant internal and external stakeholder involvement in the NRC’s development of NUREG-2191 and NUREG-2192. The NRC staff reviewed 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; 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.

In December 2017, the NRC staff also 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 NUREG-2191 and NUREG-2192 by making the necessary revisions to the existing license renewal guidance documents for 60 years of operation (i.e., NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” Revision 2, and NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” Revision 2, both issued in December 2010). These revisions accounted for expected aging management needs for the 60- to 80-year operating period, new operating experience, and additional lessons learned and incorporated license renewal interim staff guidance (LR-ISG) documents. LR-ISG documents are defined as guidance documents that the NRC issues for use by applicants until the guidance is incorporated into the next formal update of documents.

The use of these guidance documents 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. The NRC continues to evaluate and update the guidance as circumstances warrant. Since it issued the last National Report, the NRC has published four subsequent license renewal interim staff guidance (SLR-ISG) documents:

- SLR-ISG-2021-01-PWRVI, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized-Water Reactors,” dated January 2021

- SLR-ISG-2021-02-MECHANICAL, “Updated Aging Management Criteria for Mechanical Portions of Subsequent License Renewal Guidance,” dated February 2021

- SLR-ISG-2021-03-STRUCTURES, “Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance,” dated February 2021

- SLR-ISG-2021-04-ELECTRICAL, “Updated Aging Management Criteria for Electrical Portions of Subsequent License Renewal Guidance,” dated February 2021

Using lessons learned from reviewing initial license renewal applications and three subsequent license renewal applications, the NRC staff aims to complete the safety 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 subsequent renewed license can request a hearing or 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. The 18-month timeline is also contingent on the NRC resources available to support the quantity of license renewal applications submitted to the NRC at any given time.

Section 14.1.4.3 of this report describes the subsequent license renewal activities in more detail.

As of August 2022, 84 of the 92 currently operating U.S. nuclear reactors have received initial license extensions, with six of these having also received subsequent renewed licenses. These six units are 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.

In February 2022, the Commission issued a decision stating that further environmental review is required for subsequent license renewal applications and it directed the staff to propose a rulemaking plan to revise NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants,” and 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” to address these issues. Based on the Commission’s direction, the staff submitted its proposal in SECY-22-0024, “Rulemaking Plan for Renewing Nuclear Power Plant Operating Licenses—Environmental Review,” dated March 25, 2022. The Commission approved the staff’s proposal on April 5, 2022.

The NRC is currently reviewing subsequent license renewal applications for North Anna Power Station, Units 1 and 2; Point Beach Nuclear Plant, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3; and St. Lucie Plant, Units 1 and 2.

2.3.1.10 Transformation at the NRC

NRC regulations provide a strong framework to ensure safety and security. However, the regulations support the current operating fleet of large light-water reactors and do not specifically account for nonlight-water technologies. The NRC wants to ensure that its regulatory framework does not present a barrier to safety enhancements and innovation.

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 transformation team submitted SECY-18-0060, “Achieving Modern Risk-Informed Regulation,” to the Commission on May 23, 2018; however, this SECY paper was later rescinded as it was overtaken by other transformation activities.

In October 2018, the NRC began the Futures Assessment effort to ensure the agency continues to meet its mission effectively in a dynamic and evolving future. The Futures Assessment effort used a scenario planning approach to understand the various ways the NRC’s external environment could change, how the NRC could be affected, and steps that the NRC could take to be prepared. “The Dynamic Futures for NRC Mission Areas” report, dated January 2019, describes the results and four hypothetical future scenarios in which the NRC might operate in 2030 and beyond. NRC employees and external stakeholders provided insights on how each of the future scenarios might impact the NRC’s future mission delivery, operations, and people. Having a better idea of potential future scenarios allows the NRC to make better short-term and long-term adjustments, to be increasingly efficient in its work, and to make better decisions.

Building on the Futures Assessment, the NRC sought to engage staff in a strategic conversation about how to best plan and prepare for the future. The conversation was intended to build internal engagement and tap the collective wisdom of the staff to help shape the NRC’s transformation strategy. It was important that the agencywide conversation be interactive and open to all NRC employees, regardless of their geographic location. Through research, the staff found that hosting a Web-based social networking event, called the Futures Jam, would be a unique way to engage with the staff in a virtual environment because it allows for wide-scale participation while incorporating a dynamic two-way communication and collaboration. The NRC Futures Jam took place June 18–20, 2019. Over 3 days, 73 percent of the NRC staff signed on to the virtual Jam platform. Over 30 percent of the staff actively participated by posting comments or liking comments from their colleagues. Using real-time analytics, active facilitation, and subsequent data analysis, the staff mined the over 4,000 comments for themes that became the foundation for the NRC's transformation strategy.

The NRC used the insights from the Futures Assessment and Futures Jam to identify four focus areas for achieving the NRC's transformation vision:

(1) Our People: We will maintain an engaged and highly skilled workforce now and in the future.

(2) Be riskSMART: We will make sound decisions while accepting well-managed risks in decisionmaking.

(3) Using Technology: We will use technology to work smarter, including using data analytics to highlight areas for regulatory attention and improvement.

(4) Innovation: We will be innovators who make timely decisions that take into account different viewpoints and fully explored options.

Each of the four focus areas was supported by one or more transformation initiatives. The initiatives were specific projects, each led by a team of NRC staff, that supported the transformation vision and focus areas. All but one of these initiatives, the Agency Desired Culture, are now complete. The NRC staff provided updates on progress in implementing the transformation initiatives during Commission meetings on October 29, 2019; September 17, 2020; and June 22, 2021. The agency is now focused on using the foundation built during its transformation to continue to modernize and sustain innovation and progress.

2.3.2 Current Safety and Regulatory Issues

The NRC and its licensees are evaluating and resolving the following potential safety and regulatory issues, presented in alphabetical order:

- advanced reactors
- construction activities at Vogtle Electric Generating Plant, Units 3 and 4
- data analytics
- licensing and oversight of analog to digital upgrades
- oversight of test reactor fuel event and restart activities
- pandemic response
- risk-informed and performance-based regulations

2.3.2.1 Advanced Reactors

Ensuring Regulatory Readiness. To review and regulate advanced nuclear reactor technology, the NRC staff developed the report "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," dated December 2016. To achieve the goals and objectives stated in the report, the NRC staff developed the "NRC Non-Light Water

Reactor Near-Term Implementation Action Plans," and "NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans," both dated July 2017. The NRC staff has made significant progress over the last few years on its activities to support the licensing of advanced reactors. These activities are consistent with the requirements of Section 103 of the Nuclear Energy Innovation and Modernization Act (NEIMA), which was signed into law on January 14, 2019.[5] The term "advanced nuclear reactor" as defined by NEIMA means a nuclear fission or fusion reactor, including a prototype plant with significant improvements in safety and reliability compared to Generation III+ commercial nuclear reactors. Advanced reactors can encompass a broad spectrum of technologies, but in this context, the NRC has focused on regulation and oversight of nonlight-water technologies intended for use as commercial nuclear power plants producing electricity or processing heat and on nonlight-water research, test, and prototype facilities.

Consistent with the requirements of NEIMA, the NRC staff is developing a risk-informed, technology-inclusive regulatory framework for optional use by applicants for new commercial advanced nuclear reactor licenses, which it plans to establish by July 31, 2025. By prioritizing rulemaking, the NRC significantly improves its readiness to establish a transformative, clear, reliable, yet appropriately flexible framework with regulations encompassing various attributes of advanced reactor technologies. This rulemaking would create 10 CFR Part 53, "Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors." The 10 CFR Part 53 framework recognizes technological advances in reactor design and allows credit in the form of operational flexibilities when an advanced reactor design can show increased margins of safety, including slower transient response times and relatively small and slow release of fission products in accident scenarios. The 10 CFR Part 53 rulemaking leverages the transformative methodology commonly known as the Licensing Modernization Project, which is described in NEI 18-04, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1, dated August 2019. The NRC endorsed the Licensing Modernization Project approach as an acceptable methodology for reviewing novel nonlight-water technologies in RG 1.233, "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," Revision 0, dated June 2020. The Licensing Modernization Project methodology focuses on key areas of the design and licensing of advanced reactors, such as the selection of licensing basis events, classification of SSCs, and assessment of defense-in-depth. The 10 CFR Part 53 rulemaking also includes alternative options for applicants to use a probabilistic risk assessment in a traditional confirmatory analysis or a bounding event analysis.

As discussed in Sections 2.3.1.7 and 2.3.3.6 of this report, as part of the 10 CFR Part 53 rulemaking, the NRC is creating a transformative security framework for advanced reactors and conducting additional rulemakings on emergency preparedness and physical security. The rulemaking on emergency preparedness for small modular reactors and other new reactor technologies would amend the NRC's regulations to add emergency preparedness requirements that are appropriate for such facilities. The rule would create a new subsection, 10 CFR 50.160, "Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," which would adopt a scalable plume exposure pathway emergency planning zone approach and a performance-based, risk-informed, consequence-oriented, and technology-inclusive emergency preparedness framework.

5 See https://www.congress.gov/bill/115th-congress/senate-bill/512/text.

The NRC is applying a graded approach to a comprehensive range of security areas, including physical security, fitness for duty, access authorization, and cybersecurity, commensurate with the risk to public health and safety. For example, the physical security rulemaking would establish voluntary alternative physical security requirements and opportunities to credit security by design under the existing regulatory framework, commensurate with the potential consequences to public health and safety and the common defense and security.

The NRC has also enhanced its advanced reactor technical readiness by developing proof-of-concept reference plant models for plant systems and accident progression and source term analysis, updating regulatory guidance, and working on endorsements of consensus codes and standards. The NRC has also initiated a project to develop a framework document for an advanced reactor construction inspection and oversight program. As part of its efforts to develop guidance on the content of applications, in 2021, the NRC staff participated in exercises with the industry to demonstrate the use of the guidance to develop portions of the license applications for the Westinghouse Electric Company eVinci microreactor design; the TerraPower Molten Chloride Reactor Experiment design; the X-energy LLC (X-energy) Xe-100 high-temperature, gas-cooled reactor design; and the Versatile Test Reactor sodium-cooled, metallic-fueled, pool-type fast reactor design. These activities are an important part of (1) the NRC's development and implementation of strategies for increased use of risk-informed, performance-based licensing evaluation techniques and guidance, and (2) the 10 CFR Part 53 rulemaking to establish a technology-inclusive regulatory framework for advanced nuclear reactors.

The NRC is also developing options for a regulatory framework for fusion energy systems, as required by NEIMA. The NRC staff issued a preliminary white paper entitled "Options for Licensing and Regulating Fusion Energy Systems," dated April 2021. The NRC staff plans to use the information in this white paper to propose options on licensing and regulating fusion energy systems. The staff expects to submit its proposal for Commission review and consideration in fall 2022.

On October 6, 2020, the NRC staff issued SECY-20-0093, "Policy and Licensing Considerations Related to Micro-Reactors," (1) informing the Commission of licensing topics related to nuclear microreactors that may necessitate departures from current regulations, related guidance, and precedent, (2) identifying potential policy issues related to licensing microreactors, and (3) describing the NRC staff's approach to facilitating licensing submittals for near-term and future deployment and operation of microreactors.

The NRC released a draft white paper, "Micro-reactors Licensing Strategies," dated September 2021, outlining optional strategies for streamlining the licensing of anticipated microreactors. These strategies leverage flexibilities in existing regulations and identify options for changes to regulatory requirements that could provide additional flexibilities, to the extent permitted under Commission policy and existing laws. The strategies aim to maximize standardization and finality using design certification, standard design approval, and topical report approvals.

<u>Licensing Activities</u>. The NRC staff is engaged in preapplication interactions with numerous prospective applicants and vendors of advanced reactor technologies, some of which have formally notified the NRC of their intent to submit applications for licenses and permits for nuclear power plants in the next several years. In addition, two vendors expressed intent to submit applications for licenses for fuel fabrication facilities to produce tristructural isotropic fuel.

On March 11, 2020, Oklo Power, LLC, a subsidiary of Oklo Inc., submitted a combined license application for its Aurora microreactor design, proposed to be constructed and operated at the Idaho National Laboratory, located in Idaho Falls, ID. This was the first combined license application for a nonlight-water reactor design submitted to the NRC. The design uses metallic fuel to produce approximately 1.5 megawatts electric (MWe) power. As part of its review of the application, the NRC staff identified that it needed additional technical information on the maximum credible accident and the safety classification of SSCs. On January 6, 2022, the NRC denied without prejudice Oklo Power, LLC's, combined license application based on its failure to provide information on these key topics. Oklo may submit a revised application in the future, if it chooses.

Kairos Power, LLC, submitted its "Preliminary Safety Analysis Report for the Kairos Power Fluoride Salt-Cooled, High Temperature Non-Power Reactor (Hermes)," on September 29, 2021, as part of the construction permit application for a 35 MWe molten salt nonpower reactor. The NRC staff accepted the application on November 29, 2021 and has begun a detailed review. Also, there is growing interest by universities in licensing new nonpower reactors using advanced reactor technologies. For instance, the NRC is conducting preapplication activities related to molten salt (liquid fueled) and high-temperature gas nonpower reactors planned to be located on university campuses.

On April 6, 2022, TRISO-X, LLC, a subsidiary of X-energy LLC, submitted a license application for a high-assay low-enriched uranium fuel fabrication facility to produce tristructural isotropic fuel.

International Cooperation. In addition to working on national issues for advanced reactor licensing, the NRC is cooperating with international counterparts. For example, under the scope of the NRC's memorandum of cooperation with the Canadian Nuclear Safety Commission (CNSC), the NRC staff has worked with the CNSC on several cooperative reviews, advanced reactor and small modular reactor technical review approaches, and preapplication activities. In August 2021, the NRC and the CNSC publicly released their first joint report for advanced reactors, "CNSC-NRC Joint Report Concerning X Energy's Reactor Pressure Vessel Construction Code Assessment White Paper," which documents the results of their collaborative activities on the Xe-100 design.

Over the last 3 years, the NRC has built strong materials research international partnerships with the United Kingdom on the use of graphitic components, with Japan on high-temperature materials and surveillance programs, and with the Czech Republic on molten salt purity and best practices, and materials compatibility. The NRC is an active participant in the Small Modular Reactor Regulators' Forum and in NEA's Working Group on the Safety of Advanced Reactors. The NRC is also cooperating with the IAEA on its assessment of the applicability of current safety standards to advanced reactors and novel technologies.

The NRC is building an agile, sustainable program for regulating advanced reactors and is developing expertise and tools to prepare for advanced reactor licensing and oversight without imposing unnecessary regulatory burden. The NRC intends to pursue further opportunities to cooperate with international counterparts to fully leverage technical resources in the resolution of nonlight-water reactor regulatory and policy challenges.

The NRC gives the status of the agency's advanced reactor activities on its public Web site at https://www.nrc.gov/reactors/new-reactors/advanced.html.

2.3.2.2 Construction Activities at Vogtle Electric Generating Plant, Units 3 and 4

On February 10, 2012, the NRC issued the combined licenses for two AP1000 reactors at the Vogtle Electric Generating Plant, Units 3 and 4, located in Waynesboro, GA.

The Vogtle Electric Generating Plant, Units 3 and 4, project used modules made offsite and assembled into larger components that make up the nuclear units. The final major module arrived at the construction site in late 2019. The licensee, Southern Nuclear Company (Southern Nuclear), placed its final module for the Vogtle Electric Generating Plant, Unit 4, in April 2021.

As described in 10 CFR 52.103(g), under a combined license, a licensee may operate the facility after the NRC makes the finding that the acceptance criteria associated with the inspections, tests, analyses and acceptance criteria (ITAAC) are met. This finding authorizes a licensee to load fuel, conduct startup testing, and transition from construction to operations. Section 2.3.3.3 of this report provides additional details about the actions NRC has taken to prepare for the transition from construction to operation.

During construction, the licensee conducts testing and evaluation to confirm and document that it has met the ITAAC acceptance criteria. The NRC verifies that all ITAAC are successfully completed through inspections and technical reviews. To date, the NRC has completed approximately 30,000 hours of ITAAC inspections. Most of the NRC inspection findings have been of very low safety significance, with the exception of two findings identified in 2021. Those findings were associated with installation of electrical components and were of low to moderate safety significance.

The regulations in 10 CFR 52.99, “Inspection During Construction,” require licensees to submit an “all ITAAC complete” notification. This notification informs the NRC staff that the plant is ready for the NRC to authorize fuel load.

Southern Nuclear submitted this “all ITAAC complete” notification for Vogtle Electric Generating Plant, Unit 3, on July 29, 2022. The NRC completed its review of any remaining ITAAC and concluded that all ITAAC have been met. The NRC issued its 10 CFR 52.103(g) letter for Unit 3 and authorized fuel load on August 3, 2022. The licensee is expected to begin loading fuel in the third or fourth quarter of calendar year 2022. Commercial operation is projected for the fourth quarter of calendar year 2022 or the first quarter of calendary year 2023.

Southern Nuclear intends to submit this “all ITAAC complete” notification for Vogtle Electric Generating Plant, Unit 4, by April 2023. The licensee is expected to begin loading fuel in the second or third quarter of calendar year 2023 and is scheduled to start commercial operation by the third or fourth quarter of calendar year 2023.

2.3.2.3 Data Analytics

The NRC is leveraging and expanding the use of information technology tools and data analytics to better adapt to trends and new technologies and improve the NRC’s decisionmaking process and the way the agency communicates with the public, licensees, and applicants.

<u>Data Warehouse</u>. In 2019, the NRC established the Data Warehouse to create a centralized repository of data from previously siloed systems to allow for more accurate and easier data

analysis and reporting. The Data Warehouse is an on-premises system of data from authoritative sources, such as the time-reporting system, Reactor Program System, and budget execution. The Reactor Program System is a Web-based application that is designed to capture information about reactor inspection and licensing activities. The Data Warehouse extracts, transforms, and loads data for developing visualizations outside of the transactional system. All of the NRC offices and staff are now able to access the Data Warehouse to gather standardized and accurate data. The NRC expects to migrate the Data Warehouse to the Azure cloud by the third quarter of FY2022, which will decrease maintenance costs and will increase the functionality of the data analytics applications.

Mission Analytics Portal. The Mission Analytics Portal and the Mission Analytics Portal-External are applications to provide stakeholders with data to make better and faster regulatory decisions. The Mission Analytics Portal is for internal use by NRC staff and management to retrieve mission-related data and present it in an easy-to-understand format. It provides critical business analytics to enhance the NRC's ability to make risk-informed decisions about how it operates and regulates. The Mission Analytics Portal provides quicker access to information and a broader reach across the four regions and different offices. Dashboards and metrics have been developed that allow users to identify issues that require more attention, enabling staff to focus on these issues. Efficiency is improved by reducing the time spent manually gathering and validating data from different sources in preparation for meetings and other regulatory activities. Additionally, access to more data improves decisionmaking and consistency. As a service for internal users, NRC staff members can connect to the Data Warehouse and produce dashboards and analytics tools for themselves.

The vision for Mission Analytics Portal-External is that it will be used by external stakeholders, including licensees, applicants, and interested groups, to retrieve, submit, and interact with regulatory information. The Mission Analytics Portal-External will transform the way the NRC engages with external stakeholders through the use of technologies that will serve to promote openness and transparency while helping the agency become more effective and efficient. The Mission Analytics Portal-External system currently allows licensees to submit event notifications and relief requests to the NRC. New modules will continue to be developed based on stakeholder input.

Dashboards. The Office of Nuclear Reactor Regulation has developed data analytics tools and dashboards to summarize and highlight trends in the Reactor Oversight Process. These tools have been consolidated on a new Operating Experience Hub to provide a single location for staff to access operating experience information. Some of the information available include historical data on NRC inspections, licensee events, and budget metrics.

The Office of Nuclear Reactor Regulation is also using dashboards to quickly meet changing business needs and to replace monthly paper reports with electronic reports. For example, (1) the open licensing actions dashboard allows staff and management to identify review actions that may be late, (2) the resources dashboards help evaluate if the staff effort in an area is under or over budgeted and it helps balance workload between staff, (3) metrics dashboards are used to report Congressional requirements, and (4) dashboards are being used to track the use of risk-informed decisionmaking as outlined in LIC-206.

Through data analytics and dashboards, the NRC is also improving its openness, efficiency, clarity, and reliability. In 2021, the agency deployed the operating reactor analytics Web site (https://www.nrc.gov/reactors/operating/oversight/analytics.html), the accident sequence precursor dashboard (https://www.nrc.gov/about-nrc/regulatory/research/

asp.html#dashboard), and the operating experience scrams dashboard (https://www.nrc.gov/reactors/operating/ops-experience/scrams.html#dashboard) to give the public access to currently available information but in a format that is much easier to understand. These dashboards provide a centralized location, which improves data search that previously required a manual search of PDFs on the public Web site or in the NRC's Agencywide Documents Access and Management System (ADAMS).

Federal Government Commitments. Data analytics activities support the NRC's implementation of the Foundations of Evidence-Based Policymaking Act of 2018 (also known as the Evidence Act). It requires agencies to establish a governance structure around evidence-based decision and policymaking, including designating a Chief Data Officer, a Chief Statistical Officer, and an Evaluation Officer. The Evidence Act also requires Federal agencies to maintain a comprehensive inventory of datasets and make data more accessible to the public and to other agencies. Reports must be submitted to Congress and the Office of Management and Budget on various topics, including a systematic plan for using evidence (i.e., data) to identify and address policy questions, an assessment of the agency's capacity for evidence-based decisionmaking, an annual report on the program evaluations that the agency plans to conduct, and a plan to make data open and accessible to the public. These activities are supported by the infrastructure the NRC has built to aggregate data in the Data Warehouse, the data analytics capabilities established by the Mission Analytics Portal, and the effort to make data more accessible to external stakeholders through the Mission Analytics Portal-External.

External Outreach. The Office of Nuclear Reactor Regulation has partnered with other organizations in the NRC, including the Office of Nuclear Regulatory Research, to organize and lead three Data Science and Artificial Intelligence workshops. These workshops attempt to establish a common terminology within the NRC along with other government agencies, national laboratories, and the nuclear industry and to identify use cases for data science applications.

2.3.2.4 Licensing, Oversight, and Facilitation of Digital Upgrades

The staff has made significant progress on several key activities that support improved clarity and reliability of the NRC's digital instrumentation and control regulatory infrastructure, and it continues to engage stakeholders on developing and implementing ongoing improvements. Examples of these activities include the following:

- The staff improved guidance in RG 1.187, "Guidance for Implementation of 10 CFR 50.59, 'Changes, Tests, and Experiments,'" by issuing Revision 3 in June 2021. Revision 3 of RG 1.187 clarifies 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."

- Regulatory Issue Summary (RIS) 2002-22, "Clarification on Endorsement of Nuclear Energy Institute Guidance in Designing Digital Upgrades in Instrumentations and Control Systems," Supplement 1, dated May 31, 2018, clarifies the guidance for preparing and documenting qualitative assessments that can be used to evaluate the likelihood of failure of a digital modification proposed for use under 10 CFR 50.59, including the likelihood of a common-cause failure for systems of lower safety significance.

- Digital Instrumentation and Control Interim Staff Guidance-06 (DI&C-ISG-06), "Licensing Process," Revision 2, dated December 2018, discusses an alternate review process to approve digital designs earlier in the life-cycle design process. Section 2.3.3.4 of this report provides additional information related to the use of the alternate review process.

- Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Defense in Depth and Diversity To Address Common-Cause Failure Due to Latent Design Defects in Digital Safety Systems," Revision 8, dated January 2021, improves the regulatory guidance for addressing common-cause failure with a graded approach commensurate with system significance.

- A new technology-inclusive design review guide, "Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews," was issued in March 2021. The guidance supports (1) flexible regulatory review processes for nonlight-water reactors within the bounds of existing regulations, and (2) a new nonlight-water reactor regulatory framework that is risk-informed and performance-based and features the NRC staff's review efforts commensurate with the demonstrated safety performance of nonlight-water reactor technologies.

- Digital instrumentation and control Inspection Procedure (IP) 52003, "Digital Instrumentation and Control Modification Inspection," dated July 2021, supports inspections of digital instrumentation and control modifications performed with license amendments, including those using the DI&C-ISG-06 alternate review process.

As a result of these improvements, the NRC staff is preparing for the licensing review of major digital upgrades to operating plants, and design reviews of advanced reactors with modern digital instrumentation and control systems. In July 2022, the staff received an application using the enhanced regulatory infrastructure for major digital upgrades to the Turkey Point Nuclear Generating, Units 3 and 4, protection systems. An application for the Limerick Generating Station, Units 1 and 2, protection systems and control room is expected before the end of the year. The staff anticipates additional operating plants to pursue projects of this nature in subsequent years. Section 18.3.2.2 of this report provides additional information about the staff's technical review of new reactor design and construction activities related to digital instrumentation and control systems.

In parallel with these increased licensing activities, the staff will continue to implement additional improvements to the digital instrumentation and control regulatory infrastructure. The staff continues to extensively engage with external stakeholders as part of this process. Examples include (1) the NRC endorsement review of NEI 17-06, "Guidance on Using IEC 61508 SIL Certification to Support the Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Related Applications," Revision 0, dated September 2019, related to commercial-grade dedication of digital equipment by third parties, (2) NRC consideration of alternate risk approaches to address digital system common-cause failure in draft NEI 20-07, Draft B, "Guidance for Addressing Software Common Cause Failure in High Safety-Significant Safety-Related Digital I&C Systems," dated August 2021, and (3) the NRC implementation of RG updates to streamline and integrate the existing set of guidance to adopt the latest Institute of Electrical and Electronics Engineers (IEEE) code and consensus standards for digital instrumentation and control design and software development.

2.3.2.5 *Oversight of National Institute of Standards and Technology Test Reactor Fuel Event and Restart Activities*

Description of the Event and the NIST and NRC Responses. On February 3, 2021, NIST operators were performing a startup of the nonpower reactor and were increasing power from approximately 10 megawatts thermal (MWt) to 20 MWt, which is the reactor's full licensed power, after a shutdown for refueling and maintenance. During the startup, the safety system automatically shut down the reactor because detectors indicated much higher than normal radiation levels in the air leaving the reactor building through the ventilation system and exhaust stack. The operators declared an "Alert" in accordance with the NIST emergency instructions and reported the event to the NRC Headquarters Operations Center. After the reactor was shut down, the workers left the building, and operators monitored the reactor using a remote station designed for that purpose. NIST ended the event later that day because radiation levels were below the criteria in the emergency instructions.

The event had no significant radiological consequences for NIST workers, the public, or the environment. No injuries were reported. Several NIST workers who were inside the building during the event were contaminated with radioactive material and exposed to higher than normal radiation levels. These workers were decontaminated, and radiation exposures were well below regulatory limits for radiation workers. Radiation measurements near the boundary of the NIST property, about 400 meters from the reactor, showed that radiation levels were near naturally occurring levels. During the event, potential radiation doses beyond the NIST property would have been less than 1 millirem, a very small fraction of the regulatory annual public dose limit of 100 millirem.

On February 9, 2021, the NRC began a special inspection to examine NIST's response to the event. NIST submitted a report to the NRC on February 16, 2021, that described the circumstances of the event. Subsequently, NIST notified the NRC that (1) the concentration of airborne radioactive material released was slightly higher but was still a small fraction of the regulatory limits, (2) the temperature of one fuel element exceeded its safety limit, causing damage to a fuel element, and (3) NIST did not meet several operational requirements in the facility. NIST submitted a followup report on May 13, 2021, with a preliminary analysis of the cause of the event. NIST determined that the fuel element was not properly seated, causing a localized loss of cooling. As a result, a small amount of melted fuel was observed on the lower grid plate surfaces near the displaced fuel element nozzle.

The NRC issued an interim special inspection report on April 14, 2021, confirming that the NIST reactor had safely shut down and that the event did not pose a risk to public health and safety. The report also provided the results of the NRC's confirmatory calculations that verified that the dose consequences of the event were significantly below regulatory limits. The NRC issued a subsequent special inspection report on March 16, 2022, documenting completion of the inspection objectives and outlining seven apparent violations of regulatory requirements associated with the event. Subsequently, the NRC and NIST participated in an Alternative Dispute Resolution process to resolve the identified violations, and on August 1, 2022, the NRC issued a Confirmatory Order documenting corrective actions required to be implemented by NIST to preclude recurrence of the event.

NIST Reactor Restart. Because a safety limit was exceeded during the event, regulations state that the NIST reactor must not restart until authorized by the NRC. NIST submitted a request for NRC authorization to restart the reactor in October 2021. The restart request included proposed actions that the NRC staff will need to review before authorizing restart. The restart decision will be informed by several NRC activities including a technical review, inspections, and enforcement actions. The NRC assembled a team of experts to review the information provided by NIST to ensure restart readiness. The NRC staff reviewed the restart request and identified supplemental information needed to support any restart decision. To support the review of the information provided by NIST, the NRC also initiated an audit of NIST in December 2021.

NIST identified during its root cause evaluation, in part, that the technical specifications of the license governing the operation of the NIST reactor did not adequately protect the fuel from damage. Therefore, on December 23, 2021, NIST submitted a request to revise these specifications to address this root cause related to the proper placement of fuel in the core to ensure proper cooling of the fuel. On July 1, 2022, the NRC issued a license amendment to NIST that revises the technical specifications related to fuel element latch verification. This change requires NIST to perform both rotational checks and visual inspection following handling of fuel within the reactor vessel and prior to operation of the reactor.
The NRC provides additional information on the incident, including copies of the event notifications, letters, and inspection reports on its public Web site at https://www.nrc.gov/reactors/non-power/event-at-nist.html.

2.3.2.6 Pandemic Response

On January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency in response to COVID-19. Following this declaration, the NRC began taking all necessary steps to protect public health and safety, including the identification of regulatory requirements that could pose challenges to the health of the workers during the public health emergency and the areas where the staff believed that temporary flexibilities, such as exemptions, would not compromise the ability of licensees to maintain the safe and secure operation of NRC-licensed facilities.

Flexibilities for the Licensees. The NRC staff held multiple public teleconferences with stakeholders to seek information and to identify areas where requests for regulatory relief may be needed and whether expedited NRC decisions would be requested. The NRC staff then issued letters communicating the site- and situation-specific information needed to review expedited exemption requests.

The NRC's Office of Nuclear Reactor Regulation also established the NRC COVID-19 Coordination Team, which is responsible for the following:

- maintaining the status of anticipated reactor licensing and inspection activities in response to the COVID-19 public health emergency
- identifying any challenges to completing the Office of Nuclear Reactor Regulation's mission-related work, changes in priorities, or resource shifts considering the COVID-19 public health emergency
- serving as the point of contact for matters raised by the industry and members of the public pertaining to COVID-19 reactor-related issues
- facilitating meetings with reactor industry representatives on public health emergency-related matters

- identifying possible efficiencies for addressing COVID-19 public health emergency-related work such as approaches to streamline the review and approval of relief requests for sites with refuelling outages

The NRC staff established and communicated additional criteria describing the conditions under which it would expedite review of licensee requests for relaxation of, or exemption from, certain regulatory requirements. However, the agency's standard for granting such regulatory relief remains unchanged. The NRC may only grant exemptions that do not present an undue risk to public health and safety, are consistent with the common defense and security, and are authorized by law. The staff reviewed all requests for COVID-19 temporary regulatory relief on a case-by-case basis and granted the requests only if adequate controls were in place to maintain safety and security. The staff issued the letters to industry describing the criteria and conditions under which it would expedite review of licensee requests for relaxation of, or exemption from, certain regulatory requirements in seven regulatory areas:

(1) work hour controls (10 CFR Part 26, "Fitness for Duty Programs")

(2) licensed operator requalification program and medical examinations (10 CFR Part 55, "Operators' Licenses")

(3) security personnel training and qualification and force-on-force exercises (10 CFR Part 73, "Physical Protection of Plants and Materials")

(4) respirator fit testing and medical exam requirements (10 CFR Part 20, "Standards for Protection against Radiation")

(5) fire protection requirements (10 CFR 50.48, "Fire Protection")

(6) owners activities report outages activities requirement (10 CFR 50.55a, "Codes and Standards")

(7) biennial emergency preparedness exercise requirements (10 CFR Parts 30, 40, 50, 52, 70, and 72)

On November 10, 2020, the NRC issued a letter, "U.S. Nuclear Regulatory Commission Updated Planned Actions Related to Certain Requirements for Operating and Decommissioning Reactor Licensees During the COVID-2019 Public Health Emergency," to provide guidance on the continued use of expedited processes beyond December 31, 2020, for requests related to the COVID-19 public health emergency in these seven regulatory areas. Enclosures to the letter address informational needs for each of regulatory area to facilitate licensees' continued use of the NRC's expedited review process, such as providing justifications for the hardships that have resulted from the COVID-19 public health emergency and information related to the potential cumulative effects of these exemptions. Additionally, on April 30, 2021, the NRC issued a letter to provide information on the NRC's planned actions related to the requirements in 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material," during the COVID-19 public health emergency.

The NRC maintains a list of all approved COVID-19-related licensing requests issued to an operating nuclear reactor licensee. Subsequent requests for changes to the license or for exemptions from the regulations consider the effect of previously issued changes to a license and exemptions, as appropriate. The staff took a number of steps to identify areas of NRC regulations that are challenging during the public health emergency, and the areas where temporary flexibilities, such as exemptions, would not compromise the ability of licensees to maintain the safe and secure operation of NRC-licensed facilities. The staff has communicated the processes available to licensees for requesting these flexibilities in a transparent way through public communications, such as teleconferences and letters. The NRC has posted these processes, approved flexibilities, notification letters, and approved requests on its public Web site at https://www.nrc.gov/about-nrc/covid-19/.

Flexibilities for the NRC Staff and Inspectors. The NRC developed temporary staff guidance to provide the NRC staff the framework for expedited processing of COVID-19 exemption requests. The guidance documents expectations and flexibilities replace or supplement the routine exemption review processes. These expectations and flexibilities are intended to enhance the staff's efficiency in responding to the needs of licensees and the public during the COVID-19 public health emergency. This temporary staff guidance expires when the COVID-19 public health emergency ends.

For COVID-19 exemptions, the NRC issued a monthly rollup or summary notice in the *Federal Register*. Instead of issuing individual *Federal Register* notices for each exemption granted, the NRC provided a compiled listing of exemptions granted each month. This listing provided transparency and saved significant staff hours and publishing costs.

For inspection and oversight at operating reactors, the NRC successfully implemented a modified inspection approach to accomplish both onsite and remote oversight activities during the COVID-19 public health emergency. Following the initial onset, in April 2020, the NRC's Office of Nuclear Reactor Regulation provided guidance for resident inspector site presence and inspection sample completion, including the use of remote samples and maximizing telework while still maintaining some onsite presence. The agency updated this guidance as the public health emergency progressed and the situation changed. The NRC staff continued to implement the baseline inspection program and initial operator licensing examinations while taking precautions recommended by the U.S. Centers for Disease Control and Prevention to minimize exposure to COVID-19.

In 2020, NRC inspectors completed over 150,000 direct inspection hours at all 59 reactor sites in the United States. This was about 110 percent of the minimum number of inspections required in the NRC's baseline inspection program. Some inspections, such as security force-on-force inspections, radiation safety, emergency preparedness and plant outage inspections, were either delayed or cancelled. The NRC achieved reasonable assurance of safe plant operation based on onsite resident inspector presence and monitoring of plant activities in accordance with established inspection requirements as well as inspectors' discussions with plant personnel; their review of plant records; the observation of overall plant performance, including findings, performance indicators, events, and equipment performance; and satisfactory completion of inspection samples. The staff further discusses the NRC's assessment of its Reactor Oversight Process during the public health emergency in SECY-21-0038, "Reactor Oversight Process Self-Assessment for Calendar Year 2020," dated April 1, 2021.

As of November 7, 2021, the inspection program guidance has been restored to the expectations before the public health emergency. The NRC has published additional information on the guidance and flexibilities provided to the inspectors during this time on its public Web site at https://www.nrc.gov/about-nrc/covid-19/reactors/inspector-guidance.html.

Improving Internal and External Communications. The COVID-19 public health emergency underscored the importance of adapting in a dynamic environment. It served as a catalyst to accelerate innovations advancing the NRC in its journey to becoming a more modern, risk-informed regulator. For example, the NRC created a section on its public Web site (https://nrcweb.nrc.gov/about-nrc/covid-19/) to provide a centralized portal for information on the NRC's actions in response to requests for regulatory relief, opportunities for public engagement, and frequently asked questions. The NRC developed a Web-based submission portal for licensees to submit COVID-19 regulatory relief requests through the NRC's public Web site; traditionally only written submittals were acceptable for review. The NRC also used the agency's internal Nuclepedia to document, memorialize, and communicate lessons learned during the response to the public health emergency. Nuclepedia is a Wiki tool that provides a collaborative platform for learning and knowledge. Section 8.1.6.2 of this report contains additional information on Nuclepedia. Examples of these COVID-19 Nuclepedia posts include Web pages with insights on reactor licensing, reactor oversight, information technology, and teleworking.

The NRC continues to monitor the effects of the public health emergency on NRC-licensed facilities as well as actions taken in response to local conditions and will continue to take appropriate regulatory steps. To date, the COVID-19 public health emergency has not resulted in safety issues or events at any NRC-licensed facility. If the NRC identifies any facility where the impact of the COVID-19 public health emergency creates concerns about continued safe operation, the agency will take necessary steps to ensure public health and safety.

2.3.2.7 Risk-Informed and Performance-Based Regulations

As part of its transformation efforts, the NRC has taken numerous steps to be more modern and to improve risk-informed and performance-based regulations and processes.

Very Low Safety Significance Issue Resolution. Stemming from the Reactor Oversight Process enhancement project, the NRC's Office of Nuclear Reactor Regulation formed a working group to evaluate and establish means of promptly assessing and resolving low safety significance issues within existing regulatory processes and help focus staff resources on issues of greater safety significance. The Low Safety Significance Issue Resolution working group was convened to assess the resolution of low safety significance issues that arise as a result of inspection activities and proposed licensing actions. The working group considered three categories of issues: (1) issues that are within the licensing basis and are covered by existing regulatory processes, (2) issues that are outside the licensing basis and are covered by the existing backfit process, and (3) issues that require significant further research to determine their licensing basis standing.

Based on the working group's proposal, the agency updated Inspection Manual Chapter 0612, "Issue Screening," Appendix B, "Issue Screening Directions," on January 1, 2020, to create a new screening step called the VLSSIR process. The VLSSIR process is used to discontinue inspection, screening, and evaluation of an issue only involving a licensing basis question. The agency's issue screening guidance allows for an issue to be dispositioned by the VLSSIR process in the following cases:

- The condition surrounding the issue of concern cannot have any potential to be greater than very low significance (i.e., not greater than green if the issue was determined to be a finding evaluated using the significance determination process).

- The inspection staff has not been able to conclude that the issue of concern is a violation or failure to meet a licensee standard.

- The resources required to resolve the current licensing basis question would not effectively and efficiently serve the agency’s mission.

The NRC’s inspection reports document issues addressed using the VLSSIR process.

Risk-Informed Process for Evaluations. The Low Safety Significance Issue Resolution working group also developed a proposal, referred to as the Risk-Informed Process for Evaluations (RIPE), to achieve a more efficient review of low safety significance license amendments and exemption requests. RIPE leverages previous risk-informed initiatives to support the evaluation of regulatory issues consistent with the key principles of integrated decisionmaking in RG 1.174, Revision 3. Using those principles, the NRC can ensure that the level of effort of the staff’s review is commensurate with the issue’s safety significance. The working group’s proposal was approved and subsequently expanded to apply to additional licensees.

To implement RIPE, as expanded, licensees must have adopted (1) TSTF-505 or TSTF-425, and (2) 10 CFR 50.69, or a RIPE integrated decisionmaking panel, as documented in NEI guidance, “NEI Guidelines for the Implementation of the Risk-Informed Process for Evaluations Integrated Decision-Making Panel,” dated August 2020.

If a licensee elects to use RIPE, the licensee will characterize the safety significance associated with the proposed license amendment or exemption request using the NRC’s “Guidelines for Characterizing the Safety Impact of Issues,” dated January 2021, and then submit its request to the NRC. If the conditions described in the RIPE guidance are met, the NRC staff will review the request using the streamlined process outlined in the Office of Nuclear Reactor Regulation Temporary Staff Guidance (TSG-DORL-2021-01), “Risk-Informed Process for Evaluations,” dated January 5, 2021.

Risk-Informed Technical Specifications. The NRC staff continues to work on initiatives to add a risk-informed component to the standard technical specifications. The NRC is reviewing licensing applications involving the establishment of a risk management approach for certain limiting conditions for operation contained within technical specifications under TSTF-505, Revision 2. 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 has also approved risk-informed changes to surveillance requirement frequencies in technical specifications under TSTF-425, Revision 3. These initiatives are intended to maintain and improve safety by incorporating risk assessment and management techniques in the technical specifications while reducing unnecessary burden.

Be riskSMART. The Be riskSMART framework supports the NRC’s focus on applying risk in decisionmaking by providing a systematic approach to making risk-informed decisions across disciplines. Be riskSMART combines traditional concepts, such as the risk triplet, risk management, the risk heat map, and risk appetite, into a plain-language framework that gives the staff confidence to apply and communicate risk insights for all kinds of NRC decisions, including in the technical, corporate, and legal arenas. The framework serves as an umbrella to increase consistency, awareness, and usability.

The framework is broad by design to accommodate NRC staff members who are both familiar and unfamiliar with RIDM and risk information. The framework uses plain language and provides a step-by-step structure to consider risk systematically, especially qualitative information. The framework does not replace any existing RIDM approaches, such as PRA and enterprise risk management. It does not revise any of the criteria already in place for making risk-informed decisions, such as reactor safety decisions involving the significance determination process.

The Be riskSMART framework has the following six steps:

(1) Be…clear about the problem.
(2) Spot…what can go right or wrong? What are the consequences? And how likely is it?
(3) Manage…what you can.
(4) Act…on a decision.
(5) Realize…the result.
(6) Teach…others what you learned.

The NRC has collected additional details as well as case studies of occasions when the NRC has successfully applied the Be riskSMART framework in various areas of its decisionmaking in NUREG/KM-0016, “Be riskSMART: Guidance for Integrating Risk Insights into NRC Decisions,” dated March 2021.

2.3.3 Major Regulatory Accomplishments

Since the issuance of the previous U.S. National Report in 2019, the NRC has achieved many regulatory accomplishments. The following are some of the major items:

- closure of the assessment of debris accumulation on sump performance issues
- closure of the open phase conditions in electric power systems issues
- construction oversight and transition to operation
- decommissioning rulemaking
- digital modernization activities at Waterford Steam Electric Station, Unit 3
- emergency preparedness requirements for small modular reactors rulemaking
- implementation of Fukushima lessons learned
- issuance of new and renewed licenses

2.3.3.1 Closure of the Assessment of Debris Accumulation on Sump Performance Issues

As discussed in Section 2.3.1.2 of this report, in 2019, the NRC closed this issue for all BWRs.

On July 23, 2019, the NRC closed out GSI-191. Even though GSI-191 is closed, each plant must still respond to GL 2004-02. GL 2004-02 includes questions about debris-related challenges to sump strainer performance, and questions about the effects of debris that passes through the strainer and reaches the core (i.e., in-vessel debris).

On September 4, 2019, the NRC staff issued a document to guide its review of licensee submittals of in-vessel issues. This guidance describes a new risk-informed method for evaluating the in-vessel issue commensurate with its safety significance for each plant configuration.

To date, 18 PWRs have provided responses using the new in-vessel guidance. GL 2004-02 has been closed for 20 PWRs—including the low-fiber plants using earlier guidance and the plants using the new in-vessel guidance. The NRC staff is currently reviewing the responses from 9 PWRs, and 5 PWRs have not submitted their final evaluations.

2.3.3.2 Closure of the Open Phase Conditions in Electric Power Systems Issues

As discussed in Section 2.3.1.6 of this report, all operating plants that are susceptible to open phase conditions have implemented plant modifications, either by submitting a license amendment request or implementing the voluntary industry initiative, to address the open phase vulnerabilities. The NRC staff is making good progress with closing out NRC Bulletin 2012-01 for the sites it inspected, as it has issued bulletin closure letters for approximately 95 percent of the operating plants. The NRC has completed revising the Reactor Oversight Process inspection procedures and Inspection Manual Chapters to provide periodic oversight of licensees' implementation of the voluntary industry initiative to address the open phase vulnerabilities.

2.3.3.3 Construction Oversight and Transition to Operation

The goal of the NRC's construction oversight program is to ensure that new nuclear power plants will operate safely. The NRC's oversight program provides reasonable assurance that licensees and their vendors detect and correct problems that could impact quality or safety. 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, describes the development of the construction inspection program for new reactor construction. Inspection Manual Chapter 2506, "Construction Reactor Oversight Process General Guidance and Basis Document," dated November 25, 2020, provides an implementation guide for the program.

The NRC designed the construction inspection program and construction assessment process to reflect the rapidly changing nature of a construction environment. The program is based upon lessons learned described in NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants," dated May 1984. Those lessons were taken from years of experience with domestic construction and from the NRC's international regulatory counterparts. 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.

To ensure its readiness to transition new reactors from construction to operations, the NRC established the Vogtle Readiness Group in 2018. The group provides oversight and management direction to NRC staff to ensure that the Vogtle Electric Generating Plant units under construction meet regulatory requirements and are safe to operate. The group is led by the senior executives with direct responsibility for the project, including the NRC's program office at headquarters and the region-based inspection organization. The Vogtle Readiness Group serves as the focal point for project status and for coordination through commercial operation of the Vogtle Electric Generating Plant, Units 3 and 4. The group also serves as the hub for communications with the Commission, the NRC's offices that support the project, 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 ITAAC closure). The integrated project plan lists the regulatory milestones for the transition of Vogtle Electric Generating Plant'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.

In October 2019, the NRC also established the Vogtle Project Office in the Office of Nuclear Reactor Regulation. The Vogtle Project Office is directly responsible for licensing and project management, and collaborates closely with the NRC's Division of Construction Oversight on oversight and inspection of ITAAC for the construction and startup. As discussed in Section 18.1.3 of this report, the Vogtle Project Office staff has been instrumental in coordinating activities with NRC's inspection staff, ensuring that licensing actions are addressed in a timely manner, coordinating tabletop exercises, communicating with the public, and addressing the complexity of the construction activities, among other tasks.

In August 2020, the NRC developed a plan, "Transition to Reactor Oversight Process for Vogtle Electric Generating Plant, Units 3 and 4," to provide an effective and efficient transition from the Construction Reactor Oversight Process to the Reactor Oversight Process. Implementation of the plan has brought about greater management awareness of construction status and has resulted in better coordination between the offices responsible for transition from construction to operations oversight. It has led to refinements in the process to conduct the transition, enabled staff to develop updated programmatic and inspection documents based on clarifications developed through the Vogtle Readiness Group, and allowed the staff to focus its inspection resources more efficiently. Furthermore, it has provided a forum through which the NRC held several public meetings for expanded stakeholder understanding of the NRC's construction oversight program.

In implementing the Construction Reactor Oversight Process, the NRC conducted approximately 50,000 hours of construction inspection at Vogtle Electric Generating Plant, Units 3 and 4. The majority of the issues identified as part of the NRC's Construction Inspection Program were determined to be of very low safety significance. However, in late 2020, Southern Nuclear, the licensee holder for the Vogtle Electric Generating Plant construction project, identified issues with the installation of Class 1E safety-related electrical cables and raceways. In response, the NRC performed a special inspection and identified two findings of low to moderate safety significance (i.e., white findings). These were the first and only NRC findings of greater-than-green significance identified at the Vogtle Electric Generating Plant, Units 3 and 4. The first of the two findings was for Southern Nuclear's failure to promptly identify and correct the installation issues. The second finding was for Southern Nuclear's failure to adequately follow design specifications when installing certain Class 1E cables. The NRC completed a final supplemental inspection in March 2022. In its inspection report dated April 19, 2022, the NRC concluded that Southern Nuclear's corrective actions were adequate, and the two white findings were closed. The NRC has posted its construction inspection reports on the public Web site at https://www.nrc.gov/reactors/new-reactors/oversight/crop/con-inspection-reports.html.

As discussed in Section 2.3.2.2 of this report, the NRC completed its construction inspection activities and authorized fuel load for the Vogtle Electric Generating Plant, Unit 3, on August 3, 2022. At that time, Unit 3 was successfully transitioned from the Construction Reactor Oversight Process to the Reactor Oversight Process. Unit 3 is expected to achieve commercial operation in the fourth quarter of calendar year 2022 or the first quarter of calendar year 2023. Construction continues at Unit 4, with commercial operation expected to commence in the third or fourth quarter of calendar year 2023.

2.3.3.4 Decommissioning Rulemaking

The NRC is proposing to amend its regulations for the decommissioning of production and utilization facilities. The goals of this rulemaking are to maintain a safe, effective, and efficient decommissioning process; reduce the need for license amendment requests and exemptions from existing regulations; incorporate lessons learned from the decommissioning process; and support the NRC's Principles of Good Regulation, including openness, clarity, and reliability.

On November 3, 2021, the Commission approved the publication of the proposed rule provided in SECY-18-0055, "Proposed Rule: Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," to amend agency regulations for the decommissioning process. On March 3, 2022, the proposed rule was published in the *Federal Register* (87 FR 12254) for a 180-day public comment period. The staff held public meetings during the public comment period. The details of these activities and related documents can be found on the NRC's public Web site at https://www.nrc.gov/waste/decommissioning/reg-guides-comm/regulations/reg-improv-trans-to-decom.html.

The proposed rule would align regulatory requirements with the reduction in radiological risk that occurs over time, while continuing to maintain safety and security. The rulemaking would adopt a graded approach in several areas, which is commensurate with the reduction in radiological risk at four levels of decommissioning:

(1) permanent cessation of operations and permanent removal of all fuel from the reactor vessel

(2) sufficient decay of fuel in the spent fuel pool such that it would not reach ignition temperature within 10 hours under adiabatic heatup conditions

(3) transfer of all fuel to dry storage

(4) removal of all fuel from the site

The proposed rule also addresses several regulatory and technical areas, including the following:

- emergency preparedness
- physical security
- cybersecurity
- drug and alcohol testing

- certified fuel handler definition and elimination of the shift technical advisor
- decommissioning funding assurance
- offsite and onsite financial protection requirements and indemnity agreements
- environmental considerations
- record retention requirements
- low-level waste transportation
- spent fuel management planning
- application of the backfit rule
- foreign ownership, control, or domination
- clarification of the scope of the license termination plan requirement

2.3.3.5 Digital Modernization Activities at Waterford Steam Electric Station, Unit 3

The NRC issued the digital instrumentation and controls license amendment for a core protection calculator upgrade using the new alternate review process for the Waterford Steam Electric Station, Unit 3. As discussed in Section 2.3.1.5 of this report, this new alternate review process facilitated approval of the amendment request earlier in the software life-cycle development process. The staff's review focused on the system architecture, including communication interfaces, fundamental design principles (i.e., independence, redundancy, determinism, and diversity and defense in depth), software and hardware development processes, the description of the licensee's vendor oversight plan, equipment qualification, human factors considerations, conformance with IEEE standards, and crediting self-diagnostic features to eliminate certain manual functional testing in technical specifications. The licensee installed the system in spring 2022.

2.3.3.6 Emergency Preparedness Requirements for Small Modular Reactors Rulemaking

As discussed in Section 2.3.1.7 of this report, on January 3, 2022, the staff submitted the draft final rule for Commission consideration. If approved by the Commission, the final rule would be technology inclusive and would provide existing and future light-water small modular reactors, nonlight-water reactor applicants and licensees, certain existing nonpower production and utilization facilities, and nonpower production and utilization facilities licensed after the effective date of the final rule the option to develop a performance-based emergency preparedness program, rather than using the existing deterministic emergency preparedness requirements in 10 CFR Part 50. The proposed NRC emergency preparedness requirements and implementing guidance would adopt a consequence-oriented, risk-informed, performance-based, and technology-inclusive approach.

2.3.3.7 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 spent fuel pool (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 became effective on September 9, 2019 (84 FR 39718), and is codified as 10 CFR 50.155, “Mitigation of Beyond-Design-Basis Events.”

All applicable operating power reactor licensees have implemented the safety enhancements required by the reliable hardened containment vent order. The NRC completed verification inspections in September 2021. Because this order applies only to a limited group of plants (i.e., 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 completed the seismic and flooding related inspections and hazard reevaluations for the request for information. Licensees implemented interim measures, if necessary, while performing additional evaluation of the impact of the reevaluated hazards on the sites. The staff reviewed the information provided and identified those sites where additional evaluations of impact were 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. All flooding integrated assessments, seismic PRAs, and limited-scope evaluations have been submitted and the NRC has completed its review of these assessments.

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.

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

2.3.3.8 Issuance of New and Renewed Licenses

Combined Licenses. The NRC has not issued combined licenses since the submittal of the U.S. National Report in 2019. No combined licenses were terminated since the submittal of the U.S. National Report in 2019. In the United States, there are a total of eight combined licenses at five sites.

Design Certifications. The NRC certified the Korea Hydro and Nuclear Power's APR1400 design on August 9, 2021. The NRC also approved NuScale Power LLC's 12-module small modular reactor design on August 28, 2020, and it is scheduled to be certified by rulemaking in late-2022.

Early Site Permits. The NRC approved one early site permit since the submittal of the U.S. National Report in 2019: Clinch River Nuclear Site on December 19, 2019.

Renewed Licenses. The NRC issued one renewed license since the submittal of the U.S. National Report in 2019: Seabrook Station, Unit 1, on March 12, 2019.

Subsequent Renewed Licenses. The NRC issued six subsequent renewed licenses since the submittal of the U.S. National Report in 2019: Turkey Point Nuclear Generating, Units 3 and 4, on December 4, 2019; Peach Bottom Atomic Power Station, Units 2 and 3, on March 5, 2020; and Surry Power Station, Units 1 and 2, on May 4, 2021. Sections 2.3.1.9 and 14.1.4.1 of this report provide additional information about the current status of subsequent license renewal activities.

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 actively participated in its peer review activities. The peer review of the 2019 U.S. National Report was progressing positively but was put on hold because of the COVID-19 public health emergency. The contracting parties were able to peer review each other's reports but were not able to draw conclusions because the eighth CNS review meeting, scheduled to take place in March 2020, was cancelled. Therefore, the contracting parties agreed to convene a joint eighth and ninth review meeting in

March 2023. The sections below give an update on the CNS peer review activities conducted thus far.

2.4.1.1 Items Resulting from the Contracting Parties' Peer Review

A review of the questions raised by other contracting parties on the 2019 U.S. National Report identified the following areas of interest:

- emergency preparedness and incident response
- Fukushima lessons learned
- human resources
- license renewal, subsequent license renewal, and aging management
- licensing and construction
- operating experience
- radiation protection
- Reactor Oversight Process
- risk-informed decisionmaking
- safety culture and human factors
- transformation at the NRC
- transition from operation to decommissioning

Because the eighth CNS review meeting was cancelled, the United States was not able to make a national presentation on these items. However, the NRC plans to focus on areas of interest highlighted by the peer review of the 2022 U.S. National Report during the next review meeting, which is scheduled to take place in March 2023. INPO, representing the U.S. nuclear industry, will also discuss its role in maintaining and improving nuclear safety.

The United States was a member of Country Group 1 during the last CNS review meeting—the seventh review meeting, which took place in March 2017. 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

The group participants concluded that the United States had several good performances in the last review cycle. 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

In 2017, Country Group 1 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

In 2017, Country Group 1 identified the following challenges for the United States:

- establishing the acceptance criteria for life extension beyond 60 years (discussed in Section 2.3.1.9 of this report)
- clarifying backfitting guidance and implementation (discussed in Section 2.3.1.4 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 Sections 2.3.2.2 and 2.3.3.3 of this report)

The current U.S. National Report addresses these issues in the sections mentioned above to assist the contracting parties draw conclusions on these previously identified challenges.

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.

<u>The First Principle of the Vienna Declaration on Nuclear Safety</u>. 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.

In 2019, the NRC also issued a new regulation, 10 CFR 50.155, to require licensees to develop, implement, and maintain strategies and guidelines to mitigate beyond-design-basis external events. Section 2.3.3.7 and Articles 12, 18, and 19 of this report discuss these requirements in more detail.

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 https://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. Section 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 2.3.3.7 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. The NRC regulations at 10 CFR 50.109, 70.76, 72.62, and 76.76, all titled "Backfitting"; 10 CFR Part 52; and 10 CFR 50.54(f) provide the requirements for proper justification of backfitting, changes affecting issue finality, and information requests, respectively. Sections 2.3.1.4 and 14.1.5.2 of this report present additional information about the backfit and issue finality processes.

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 6.3.4 and 6.3.10 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.

The Third Principle of the Vienna Declaration on Nuclear Safety. 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 Ninth 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 that the areas of focus for these sessions were aging management and safety culture. Because the eighth CNS review meeting was cancelled, in October 2021, the contracting parties agreed to host these sessions in the joint eighth and ninth review meeting in March 2023.

Aging Management. The NRC continues to give special focus to issues associated with aging management and license renewal. The NRC has also issued subsequent license renewals, which 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 2.3.1.9 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 10.3 of this report presents more information on safety culture.

2.4.2 Integrated Regulatory Review Service

IRRS missions help the host Member State strengthen and enhance the effectiveness of its regulatory infrastructure for nuclear, radiation, radioactive waste and transport safety. The NRC regularly provides technical experts, often at a senior leadership level, to lead or 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 OSART program assists Member States in strengthening the safety of their nuclear power plants during commissioning and operation, comparing actual practices with IAEA safety standards. The NRC regularly provides technical experts, often at a senior leadership level, to participate in OSART missions around the world. In August 2017, the 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. All findings were satisfactorily addressed, as discussed in greater detail in Section 8.1.5.3 of this report. The next OSART mission in the United States was scheduled to take place in 2020 at the Wolf Creek Generating Station. This mission was rescheduled for 2023 because of the COVID-19 pandemic.

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 https://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, "Congressional Budget Justification: Fiscal Year 2023," Volume 38, dated April 2022. 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[6] 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

6 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 2021, 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 2020 and 2019.

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, "2021–2022 Information Digest," Volume 33, dated August 2021. Since the issuance of the 2019 U.S. National Report, five reactors have ceased operations, bringing the total to 92 operating power reactors in the United States. Two additional operating reactors intend to permanently cease operations in the next few years.

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 2.3.2.2 and 2.3.3.3 of this report.

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 and evaluate the applicant's financial qualifications to operate a commercial nuclear facility. 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 power reactor under 10 CFR Part 50. However, in 2015, the NRC issued a 10 CFR Part 50 license for Watts Bar Nuclear Plant, Unit 2, completing the licensing process that began with the issuance of a construction permit in 1973.

In 1989, the NRC adopted a single-step process, which 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. Currently, the NRC has 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 May 2019, the NRC issued a direct final rule certifying the Korea Hydro and Nuclear Power APR1400. In August 2020, the NRC approved NuScale Power LLC's 12-module small modular reactor design, and a proposed rule for certification is currently in process. The Mitsubishi's U.S. Advanced Pressurized-Water Reactor (U.S. APWR) design certification application review has been suspended. In March 2020, the NRC staff completed the technical review of the General Electric-Hitachi Advanced Boiling-Water Reactor (ABWR) design certification renewal application. In December 2020, the NRC staff provided SECY-20-0112, "Direct Final Rule: Advanced Boiling Water Reactor Design Certification Renewal," to the Commission for its consideration. In July 2021, the NRC published the direct final rule in the *Federal Register* (86 FR 34905), with the companion proposed rule (86 FR 35023) for public comment. The NRC received no public comments, and in August 2021, it published a *Federal Register* notice (86 FR 44262) that confirmed the ABWR design certification renewal rule, effective in September 2021.

Additionally, on February 27, 2021, the AP1000 design certification in Appendix D, "Design Certification Rule for the AP1000 Design," to 10 CFR Part 52 expired. In June 2020, Westinghouse requested that the NRC extend the duration of the AP1000 design certification by 5 years. In response, in SRM-SECY-20-0082, "Rulemaking Plan to Extend the Duration of the AP1000 Design Certification," dated November 17, 2020, the Commission approved the staff's proposal to amend the design certification for the AP1000 standard plant design and extend the duration of the design certification for 5 years. With this approved extension, the AP1000 design certification remains valid for referencing until February 27, 2026.

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 six early site permits and two limited work authorizations that allow the permit holder to perform limited construction activities at a site. The staff has approved one early site permit for the Clinch River Nuclear Site since the issuance of the last U.S. National Report. Articles 18 and 19 of this report provide more detail about the 10 CFR Part 52 regulations.

The NRC's reactor licensing process also 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 on these items.

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 using a graded approach. 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, 2022, the Action Matrix assessment of licensee performance at nuclear reactors was as follows:

- Column 1: 91 reactor units in Licensee Response
- Column 2: 1 reactor units in Regulatory Response
- Column 3: no reactor unit in Degraded Performance
- Column 4: no units in Multiple/Repetitive Degraded Cornerstone
- Column 5: no units in Unacceptable Performance

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 results of annual Reactor Oversight Process self-assessments indicate that the program remains effective. SECY-22-0029, "Reactor Oversight Process Self-Assessment for Calendar Year 2021," dated April 8, 2022, documents the most recent status of the NRC's self-assessment program. However, 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 changes recommended from the NRC's transformation initiative. Through the transformation initiative, the NRC staff received 72 staff-submitted recommendations on the Reactor Oversight Process, and an additional 27 recommendations submitted by the 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 seek Commission approval to implement program changes to the treatment of greater-than-green inspection findings and performance indicators to provide greater incentive for power reactor licensees to complete supplemental inspections as soon as practicable. The staff will also seek Commission approval to revise the emergency preparedness significance determination process to make it more risk informed.

6.3.3 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 also 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 (CCDP) or an increase in core damage probability (ΔCDP) greater than or equal to 1×10^{-6} to be a precursor. The program defines a significant precursor as an event with a CCDP or an ΔCDP greater than or equal to 1×10^{-3}.

The latest program results, trend analyses, and insights are documented in “U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program 2021 Annual Report,” dated June 2022. This report provides the results of the Accident Sequence Precursor Program for 2021. In addition, it notes the following key insights for the past 10 years (2012 through 2021):

- There were five important precursors during this period, all due to initiating events.
- The ratio of precursors identified via independent Accident Sequence Precursor evaluations continues to decrease.
- The most frequent initiating events that resulted in precursors were loss of offsite power and losses of a condenser heat sink. Long duration loss of offsite power occurring at single-unit site have a high likelihood of resulting in a higher-risk precursors.
- Natural phenomena caused 11 precursor. Snow, ice, and lightning were the most frequent causes.
- The most frequent SSC failures observed in precursors were associated to emergency diesel generators, flood protection, and switchyard.
- There are no indications of increasing risk due to the potential of cumulative impact of risk-informed initiatives.
- No new component failure modes or mechanisms were identified.
- The likelihood and impacts of accident sequences have not changed.

6.3.4 Operating Experience Program

The NRC recognizes that the effective use of operating experience is important for the agency’s safety mission. Under the current NRC Strategic Plan, the agency is committed to using lessons learned from domestic and international operating experience and other sources as part of its effort to achieve the goal of safety. As a result, the NRC’s emphasis on the effective use of operating experience remains strong.

The fundamental aim of the Operating Experience Program, described in more detail in Sections 18.4 and 19.7 of this report, 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 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.

As discussed in Section 2.3.2.3 of this report, the NRC uses data analytics to assist in the evaluation and communication of information. Visualizations of trends and inputs help identify issues requiring additional analysis and point to areas that could benefit from additional inspection. Through the Operating Experience Program, the NRC has compiled a variety of graphics, including links to source material, to allow NRC staff to perform data searches, filter information relevant to ongoing reviews, and better understand how individual events fit into the broader context of overall risk exposure. Section 19.7 of this report discusses operating experience in more detail.

The agency's public Web site at https://www.nrc.gov/reading-rm/doc-collections/event-status/index.html contains all of the event reports that licensees have submitted to the NRC.

6.3.5 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 https://www.nrc.gov/about-nrc/regulatory/gen-issues.html; a history of generic issues appears in NUREG-0933, "Resolution of Generic Safety Issues." To date, the NRC has issued 35 supplements, which include instructions for incorporating the revised pages of the document.

6.3.6 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. Rulemaking authority for the NRC is vested in the Commission. The Commission has delegated authority for some categories of rulemakings to the NRC's Executive Director for Operations. For example, the Executive Director for Operations has been delegated authority for rulemakings that are minor, corrective, or nonpolicy in nature. The Commission may also delegate individual rulemakings to the NRC staff. The NRC may pursue a rulemaking based on a congressional mandate, an Executive Order, a petition for rulemaking from outside the NRC, Commission direction, 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 rulemaking plan before initiating a new rulemaking activity that requires Commission approval. The Commission reviews this plan and issues its decision (e.g., approval or denial) on the new rulemaking activity. The Commission can approve a rulemaking plan with modifications or deny it with additional direction to the staff, for example to revise and then resubmit the rulemaking plan based on a different approach. The staff can request that the Commission delegate the rulemaking or any stages thereof to the staff such that further interaction with the Commission is not required unless changes to the rulemaking plan are needed. The staff may also ask the Commission to approve discontinuing or delaying a rulemaking activity 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 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 and may include specific questions related to the petition. The NRC staff evaluates the petition and any comments received and submits a plan for rulemaking or a petition denial for consideration by the Commission. The NRC 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 30 to 60 days to provide written comments for consideration, but longer comment periods of 75 to 90 days are often provided. The NRC can also extend comment periods after publication, if appropriate. For example, the NRC extended several rulemakings, as requested by members of the public, to account for difficulties presented by the COVID-19 public health emergency. Generally, all rules are issued for public comment. Those rules exempted from the requirement for public comment are rules for which delaying their publication to receive comments would be contrary to public interest, unnecessary, or impracticable. Although an opportunity for comment is not required, the NRC has discretion to afford an opportunity for comment on these rules. 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, if required, and publication in the *Federal Register*.

In addition to rulemakings that issue or amend regulations (also known as "legislative rules"), statutory requirements mandate that certain other documents that generically address the public or regulated entities follow the same rulemaking procedures. The NRC has many forms of these nonlegislative rules (sometimes included under the umbrella term "guidance") that the NRC issues in accordance with the NRC's rulemaking requirements in 10 CFR 2.804, "Notice of Proposed Rulemaking." Specifically, 10 CFR 2.804 provides requirements for issuance of rules of agency organization, procedure, or practice, interpretive rules (interpretations of regulation or statute), and general statements of policy. An opportunity for comment is not required for rules of agency organization, procedure, or practice, and interpretive rules or general statements of policy may sometimes use a post-promulgation comment process. Post-promulgation comment means that the document is issued as final, but the NRC formally invites and responds to public comments and may make changes to the interpretive rule or general statement of policy, if appropriate.

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 https://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 ADAMS. The NRC also ensures that all documents related to rulemakings are available in the NRC's Public Document Room.

Once approved by the Commission or authorized NRC staff official, 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.

6.3.7 Fire Protection Regulation Program

To support the implementation of 10 CFR 50.48(c), the NRC issued RG 1.205, Revision 2, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," in May 2021. This document reflects lessons learned from the pilot application reviews and supports the licensees that have adopted the rule. In June 2010, the NRC approved the first risk-informed fire protection program for Shearon Harris Nuclear Power Plant. The agency has approved all the risk-informed fire protection program applications that it received, and all transitions have been completed. This represents 46 currently operating reactors. Nuclear power plants that have not transitioned to the risk-informed, performance-based fire protection rule are regulated under their current, deterministic licensing bases.

The NRC also developed combined guidance to conduct fire protection team inspections. IP 71111.21N.05, "Fire Protection Team Inspection (FPTI)," dated January 1, 2020, combined earlier inspection procedures into a single document applicable to all plants under either regulatory framework. 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 all sites.

RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 4, dated May 2021, provides regulatory guidance for licensees on fire protection issues, including the treatment of circuit failures in response to fire damage.

The new revisions of RG 1.189 and RG 1.205 include the latest guidance on fire-induced multiple spurious operations. 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 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 mathematical fire models for regulatory use
- performing specialized fire testing on items such as electrical cables for hot shorts and fire properties of materials, including transient combustibles
- evaluating the risk posed by high energy arcing faults
- 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 NIST, EPRI, the DOE's national laboratories, and international groups such as the NEA. The NRC is currently the Operating Agent in partnership with 10 other international members for the NEA high energy arcing fault project. The NRC led and completed the first NEA high energy arcing fault experimental program and published the results as NEA/CSNI/R(2017)7, "Report on the Testing Phase (2014-2016) of the High Energy Arcing Fault Events (HEAF) Project: Experimental Results from the International Energy Arcing Fault Research Programme," dated May 2017.

6.3.8 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 10 CFR 20.1406, "Minimization of Contamination," which requires new applicants to describe how facility design and procedures will facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and the environment and the generation of radioactive waste. This requirement emphasized the importance of early planning for new applications and complemented existing requirements for applicants and licensees to have radiation protection programs aimed at reducing exposure and minimizing waste regulation. New applicants use the guidance in RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," dated June 2008, to facilitate decommissioning and minimize contamination and radioactive waste generation.

In 2011, the NRC issued the Decommissioning Planning Rule, which added a new 10 CFR 20.1406(c) and updated 10 CFR 20.1501, "General." RG 4.22, "Decommissioning Planning during Operations," dated December 2012, contains guidance for implementing the rule. Under 10 CFR 20.1406(c), licensees must, to the extent practical, conduct operations to minimize the introduction of residual radioactivity into the site, including the subsurface, in accordance with existing radiation protection requirements and the radiological criteria for license termination. 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. As discussed in RG 4.22, after identifying a leak that would require remediation to terminate the licensee, the licensee should provide additional decommissioning funding to remediate the contamination before license termination unless the licensee performs the remediation during the operational phase of the facility.

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, "Termination of License." 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 (for a Federal licensee), contractual obligation, or a combination of these methods, which are described in 10 CFR 50.75(e)(1)(i-vi). 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. Generally, electric utility licensees use external sinking funds to collect their decommissioning funds, while nonelectric utility 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 contain requirements to maintain spent fuel integrity.

The current NRC reactor decommissioning requirements have been implemented safely. Since the early 1980s, 25 power and early demonstration reactors have undergone decommissioning or are in long-term safe storage under NRC jurisdiction. Of the 25 power and early demonstration reactors in decommissioning, 12 have elected the SAFSTOR (long-term storage) option and 13 have elected the DECON (active decommissioning) option. Generally, licensees transitioning from operations to decommissioning request several license amendments and exemptions from current NRC regulations to align requirements with their decommissioning status. As discussed in Section 2.3.3.4 of this report, the Commission directed the NRC staff to conduct a rulemaking to maintain a safe, effective, and efficient decommissioning process; reduce the need for license amendment requests and exemptions from existing regulations; incorporate lessons learned from the existing decommissioning process; and support the NRC's Principles of Good Regulation, including openness, clarity, and reliability.

6.3.9 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.

6.3.10 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. Relevant examples include the following:

On December 10, 2018, the NRC issued RIS 2018-06, "Clarification of the Requirements for Reactor Pressure Vessel Upper Head Bare Metal Visual Examinations" to clarify the requirements for bare metal visual examination per the American Society of Mechanical Engineers (ASME) Code.

On September 8, 2020, the NRC issued IN 2020-01, "Increased Electronic Equipment Issues After Electrostatic Cleaning," to inform U.S. operating power reactor licensees and State radiation officers of recent operating experience associated with the use of electrostatic spray cleaning.

On June 3, 2019, the NRC issued IN 2019-02, “Emergency Diesel Generator Excitation System Diode Failures,” to inform licensees of operating experience with regard to emergency diesel generator excitation system diode failures. These diode failures may cause affected emergency diesel generators not to be able to operate for their full mission times following a loss of offsite power event.

On September 15, 2020, the NRC issued IN 2020-02, “FLEX Diesel Generator Operational Challenges,” to inform licensees of recent operational challenges involving diverse and flexible coping strategies (commonly known as FLEX) equipment at nuclear power plants. NRC licensees use this equipment to implement FLEX capability for long-term core cooling, spent fuel cooling, and containment integrity in a beyond-design-basis event scenario.

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 orders to operating power reactor licensees and construction permit holders requiring them to take critical actions.

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, contains the legal framework for the regulation of civilian nuclear installations. This act provides broad requirements, authorizations, and principles and leaves to the regulatory body (now the NRC) to address many of 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 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 <u>Provisions of the Legislative and Regulatory Framework</u>

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 issues.

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 readjudicate, 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 $326,163.

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 in 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. Section 223 of the Atomic Energy Act 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 NRC). It also explains how the functions of the NRC are separate from those of bodies responsible for promoting research, development and advancement of nuclear energy (e.g., DOE). It discusses financial and human resources aspects, the regulatory body's international responsibilities, its ethics rules, and its policy for maintaining openness and transparency.

8.1 <u>The Regulatory Body</u>

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. Through the Atomic Energy Act, the U.S. Congress established the national policy of developing the peaceful uses of atomic energy. It is this law that provides the NRC with licensing and regulatory authority over civilian radioactive materials and facilities possessing and utilizing such materials. The U.S. Congress has amended this law or enacted additional, more specialized, statutes over the years to address developing technology and changing regulatory needs. This includes the subjects of high-level radioactive waste, low-level radioactive waste, mill tailings, 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 military applications of nuclear energy, the DOE's nuclear weapons programs and facilities, or the 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.

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. Its Commissioners are appointed by the U.S. President, with the advice and consent of the U.S. Senate, to serve fixed 5-year terms, and they have statutory protection from removal. The NRC independently formulates safety and security standards, issues licenses and certifications, and conducts oversight of regulated activities, without unwarranted influence from promotional or economic considerations. Section 8.2 of this report contains more information.

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, whose members are appointed by the President and confirmed by the U.S. Senate to serve staggered 5-year terms. No more than three Commissioners can be a member of the same political party. As a collegial body, the Commission formulates policy, issues regulations governing the safe and secure use of radioactive materials, issues orders to licensees, and adjudicates legal matters brought before it. Each Commissioner has equal responsibility and equal vote in such matters.

The President designates one member to serve as Chairman, who acts as the official spokesperson and principal executive officer of the agency. Through Reorganization Plan No. 1 of 1980, Congress clarified and strengthened the executive and administrative roles of the Chairman, who is required to delegate certain day-to-day functions to the Executive Director for Operations, subject to the Chairman's direction and supervision. The Reorganization Plan also transfers to the Chairman all the functions of the Commission in the event of any emergency concerning a particular facility or materials licensed or regulated by the agency.

8.1.3.2 Component Offices of the Commission

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

- Office of the Executive Director for Operations. The Executive Director for Operations is the chief 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, which serves as a liaison with dignitaries, and the Federal and External Affairs Program, which serves as a liaison with other Federal agencies and external organizations, 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 support both the NRC's regulatory health and safety mission and the agency's internal operational activities. Since the issuance of the previous U.S. National Report, the former Office of New Reactors was consolidated into the existing Office of Nuclear Reactor Regulation. The current offices have the following roles:

- 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, 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 program for personnel and physical security.

- 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, strategic workforce planning (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 the telework program and 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 houses the Allegations Center of Expertise, which oversees the agency's Allegation Management Program and handles allegations. 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. 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 Nuclear Material Safety and Safeguards. The Office of Nuclear Material Safety and Safeguards is responsible for the licensing and regulation of facilities and materials associated with the processing, transport, and handling of nuclear materials, including uranium recovery activities and the fuel used in commercial nuclear reactors. The office performs regulatory activities that provide for the safe and secure decommissioning of reactor materials sites; the safe storage, transportation, and disposal of radioactive waste and spent nuclear fuel; and the transportation of radioactive materials regulated under the Atomic Energy Act. The office also ensures safety and security by implementing regulatory programs for licensing, inspection, and assessment of licensee performance; events analysis; enforcement; and identification and resolution of generic issues. The office implements the NRC's Agreement State Program, coordinates actions and communications with Native American Tribal governments, and supports agency rulemaking, environmental review, and financial assurance projects.

- Office of Nuclear Reactor Regulation. The Office of Nuclear Reactor Regulation is responsible for accomplishing key components of the NRC's nuclear reactor safety and security mission to protect public health and safety and the environment. To do so, the office conducts a broad range of regulatory activities in the areas of rulemaking, licensing, oversight, siting, and incident response for operating commercial nuclear power reactors, new commercial nuclear power reactors, advanced reactor technologies, and nonpower production and utilization facilities. The office also houses the EMBARK Venture Studio, which is an organization that serves as a creative catalyst to launch innovative initiatives to improve the reactor safety program.

- Office of Nuclear Regulatory Research. The Office of Nuclear Regulatory Research provides independent technical advice, tools, and information to make timely regulatory judgments, anticipating potentially significant safety problems and resolving safety issues. This includes conducting confirmatory experiments and analyses, developing technical bases that inform NRC's safety decisions, and preparing the NRC for the future by evaluating the safety aspects of new technologies and designs for advanced nuclear reactors, materials, nuclear wastes, and security. The office uses its own expertise and collaborates with partner offices at the NRC, national laboratories, the DOE and other Federal agencies, U.S. universities, and international organizations and partners. Based on research results and experience gained, the office recommends regulatory actions to resolve ongoing and potential safety issues for nuclear power plants and other facilities regulated by the NRC, including those issues designated as generic issues. The office also develops the technical basis for risk-informed, performance-based regulations in all areas regulated by the NRC.

- Office of Nuclear Security and Incident Response. The Office of Nuclear Security and Incident Response is responsible for developing overall agency policy and providing management direction for evaluation and assessment of technical issues involving security at nuclear facilities. The office is the agency's safeguards and security interface with the U.S. Department of Homeland Security (DHS), the DOE, the intelligence and law enforcement communities, and other Federal agencies. The office develops emergency preparedness policies, regulations, programs, and guidance for both currently licensed nuclear reactors and potential new nuclear reactors. The office conducts the agency's program for response to incidents and is the agency's 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, allegations, enforcement, emergency and incident response, Agreement States program activities, 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, and independent of the NRC staff. By law, all committee meetings are open to public observation, unless a specific exception allows for closure.

- 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 backfits and changes affecting issue finality are appropriately justified, based on the backfit and issue finality provisions of applicable NRC regulations and Commission 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 the Atomic Safety and Licensing Board Panel, which is a panel of administrative judges who conduct hearings and are authorized by the Commission to make initial or final decisions in adjudications concerning the granting, suspending, revoking, or amending of any NRC license or authorization. The boards are typically composed of three members: one lawyer and two technical experts. Board decisions are subject to Commission review, either on appeal by one of the parties to the adjudication or on the Commission's own motion. 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, and the prevention and detection of fraud, waste, abuse, and mismanagement. 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 include various components within the Executive Office of the President, FEMA, 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:

- <u>Executive Office of the President</u>. The Executive Office of the President, within the White House, comprises several offices and agencies that provide support for the President's policies and programs. Although the President cannot directly set NRC regulatory policy because of the NRC's status as an independent agency, the NRC may engage with components of the Executive Office of the President concerning administrative or organizational functions of the executive branch. For example, the NRC frequently interacts with the Office of Management and Budget (OMB), which is the component within the Executive Office of the President that assists the President in the preparation of the annual budget. The NRC submits its annual budget request to the OMB. Thereafter, the President submits the annual budget, including funding for the NRC, to the U.S. Congress for authorization. The OMB may also issue guidance for all executive branch agencies, including independent agencies, on matters pertaining to government operations.

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. The National Security Council, also located within the Executive Office of the President, is tasked with coordinating executive branch policies and activities concerning matters of national security and foreign policy. 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 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. 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.

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 specified nuclear materials to those States willing to assume that authority within its areas of jurisdiction. 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-nine States have signed formal agreements with the NRC and have assumed regulatory responsibility over certain byproduct, source, and small quantities of special nuclear materials. Two states have also sent letters of intent to seek agreements with the NRC. Agreement States receive no Federal funds to support the operations of their regulatory programs. However, the NRC does provide technical training to Agreement State staff to ensure a consistent and robust National Materials Program. The NRC conducts periodic 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 concerning nuclear safety or NRC operations. As noted above, the U.S. Senate also votes on whether to confirms the President's 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.

- <u>Senate Oversight</u>. 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, Climate, 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.

- <u>House Oversight</u>. 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.

- <u>Other Relevant Committees</u>. 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 engagement is 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, 2014 (79 FR 39415). NRC international activities also support U.S. foreign policy objectives related to nonproliferation and the safe and secure use of nuclear materials. The NRC 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 globally to help countries develop effective regulatory programs and rigorous safety and security standards. Some multilateral activities are carried out under the auspices of the IAEA, the NEA, or other international organizations. The NRC conducts other activities directly with counterparts under bilateral technical information exchange cooperation arrangements. The NRC's "International Strategy 2021–2025" brochure, dated August 2021, contains more detailed information about the strategic objectives of the NRC's international engagements.

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. The NRC staff regularly participates 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 universal ratification and implementation.

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 ensuring 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 matters that fall within these agencies' jurisdictions.

International Organizations and Associations. The NRC actively participates in a broad scope of programs of the two major international nuclear energy organizations, the IAEA and the 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 by the NRC or U.S. 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, not including the period during which travel was restricted because of the COVID-19 pandemic, the NRC supports 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 standing technical 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 multilateral work with the IAEA and the NEA, as well as on a bilateral basis, to support countries seeking to enhance their nuclear regulatory programs. The NRC staff contributes to 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 regulatory agencies in more than 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 develop their respective regulatory capabilities and nuclear safety infrastructure for oversight of a nuclear power reactor, research reactor, or radioactive materials program. The NRC also hosts staff from international counterparts for short-term assignments to enhance regulatory information sharing and provides opportunities for its international counterparts to participate in NRC-sponsored virtual and in-person training. 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 on average, 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 initially assisted 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 issues related to new reactors, aging management of existing nuclear facilities, 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 on regulatory development (codes and standards, fundamentals of reactor regulation and safety, PRA, and quality assurance), agency infrastructure development (organizational planning and safety culture), regulatory licensing development (construction permit application review and site application review), and regulatory oversight 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 continues 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. As a result, in 2020–2021, the NRC published 19 new or revised RGs that harmonize with or reference IAEA safety standards.

8.1.5.2 Integrated Regulatory Review Service Mission

IRRS missions are part of an IAEA program that helps the host Member States strengthen and enhance the effectiveness of their regulatory infrastructure for nuclear, radiation, radioactive waste, and transport safety. 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 OSART program assists Member States in strengthening the safety of their nuclear power plants during commissioning and operation, comparing actual practices with IAEA safety standards. 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 the OSART program. 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 followup visit in April 2019. The followup team concluded that all recommendations and suggestions were either resolved or categorized as areas with significant satisfactory progress.

The next OSART mission in the United States was scheduled to take place in 2020 at the Wolf Creek Generation Station in Kansas. This mission was rescheduled for 2023 because of the pandemic.

8.1.6 Financial and Human Resources

8.1.6.1 Financial Resources

As of October 1, 2021, 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 2022 total budget authority was $903.7 million ($887.7 million enacted budget and $16 million authorized carryover), including the budget for the Office of the Inspector General.

The NRC FY 2022 budget was financed with $756.7 million from user fees and $131.0 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 the disabled; employing veterans; enhancing diversity, which includes focusing on women in technical positions; 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 in which knowledge is continually acquired, shared, and applied to enhance individual, team or organizational performance. Such a culture supports the NRC's objective of sustaining a learning environment that fosters continuing improvement in performance to meet the agency's mission through knowledge management, training, coaching, and mentoring. The NRC has formal training programs that focus on technical, leadership, and professional development training. The technical training programs support the agency's qualification programs and provide the technical knowledge, skills and competencies for the various disciplines the agency needs in its reactor, materials, and security programs. The leadership training program, the NRC's Leaders Academy, has a broad suite of competency-based training for staff at all levels. At the lower grades, for early-career or junior staff, there is a Leader At All Levels certificate program, followed by an Aspiring Leaders certificate program for mid-career staff. For supervisors, there is a supervisory development program, and for staff aspiring to be senior executives, there is a Senior Executive Service Candidate Development Program, which supports the success planning process. Additionally, the professional training program fosters career development for staff with self-paced, virtual, and instructor-led courses.

The NRC has operationalized blended learning strategies that combine educational techniques to optimize course delivery. 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. With blended learning that includes on-demand and self-directed learning, employees have the opportunity to take more ownership of their training. Benefits of incorporating blended learning also include the ability for learners to gain or improve knowledge at any time and incorporate skills practice on the job, while saving the agency money by reducing travel costs associated with training attendance and improving staff productivity by reducing time away from work.

The agency has invested in a competency model initiative, and, to date, approximately half of the agency staff members have a competency model assigned to them. The models are a tool to improve the job skills (i.e., competencies) of today's workforce. The competency models provide personnel with the high-level tasks for their jobs and when assessed against those tasks, staff members and their supervisors can engage in developmental discussions. The employee is armed with information from the assessment and can plot their learning path to bridge a skill gap or broaden a skill in their current job. These meaningful discussions between supervisor and employee can lead to powerful dialogues on the developmental needs for the individual, whether it is about addressing a competency or skills gap or enhancing a skill at higher proficiency levels.

The NRC culture values formal training as provided through the technical, professional, and leadership training. However, the culture also values learning that is provided through other effective means such as mentoring. Senior staff or highly skilled staff with expertise and experience provide mentoring to an employee in need of developing a competency or skill. This mentoring mindset focuses on helping the employee perform successfully in a job.

Another learning vehicle is knowledge management, which remains a top priority and is an integrated part of training and the agency's Strategic Plan. The agency has established formal and informal programs to ensure that the NRC captures and preserves knowledge to assist with employee development and organizational performance. The agency recently implemented a Wiki tool, commonly known as Nuclepedia, that provides a collaborative platform for learning and knowledge. Staff members are encouraged to capture their learning in Wiki pages and to reference the tool for any formal documents, job aids, webinars, and knowledge management events. This Wiki, as it grows and multiplies, is expected to provide a valuable source of information for technical and corporate information, which will enhance informal learning for staff. Multiple contributing activities capture the formal knowledge management program:

- creating communities of practice that enable the sharing of relevant knowledge and critical skills among employees who perform the same job function
- capturing operating experience; new information on safety and security issues; and knowledge gained from inspection, research, and licensing activities in regulatory guidance
- capturing relevant critical knowledge from employees departing the agency
- creating information sites (SharePoint and Wiki) that are a go-to source of information for staff

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 strategic workforce plans, including the current and future knowledge management needs of the agency, are met. Participation by every office in the agency, in this system of governance, enhances knowledge management operations and strategies.

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 2021, more than 61,000 public users accessed ADAMS more than 103,000 times, and users requested documents more than 23.6 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 records, unless a specific exemption applies.

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 September 2021, 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. In 2021, the NRC's Web site had more than 3.1 million individual visitors. The Web site was visited approximately 5 million times, and visitors requested pages more than 54.7 million times. 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 August 2022, the NRC had more than 15,000 Twitter followers. The agency has sent a total of 5,000 tweets—an average of 42 per month since the launch of the platform.

The agency launched its Facebook page in August 2014. Since that time, its page has gained more than 10,530 follows and 383,000 engagements on more than 1,800 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 300 publicly available videos, which have received nearly 4575,000 views. More than 2,800 users subscribe to the NRC's YouTube channel and are notified each time new content is posted. Since February 2012, the NRC has published about 4,300 photos and graphics to its Flickr account, which have been collectively viewed approximately 9.8 million times.

The NRC also leverages the LinkedIn platform both for recruitment-focused information and as a complement to the agency's Facebook page. The NRC started using LinkedIn in 2014. Since we began tracking in March 2021, the agency has contributed 290 posts to the NRC's LinkedIn page. Those posts earned more than 62,000 engagements. In that time the page added nearly 4,800 new followers.

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 updated its public meeting policies in March 2021, redefining meeting categories, describing the NRC's expectations for respectful, civil discussions during public meetings, and reiterating the agency's commitment to transparency (86 FR 14964).

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. In 2021 and 2022, the NRC leveraged videoconferencing and remote participation tools to provide a fully virtual Regulatory Information Conference during the global pandemic.

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 Independence of the Regulatory Body and Separation of Functions from Those Promoting Nuclear Energy

Legislation enacted by the U.S. Congress ensures the effective independence of the NRC and the separation of its functions from those of any other body concerned with the promotion or utilization of nuclear energy. Originally, the regulatory and promotional responsibilities for nuclear energy in the United States were combined in a single 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. Section 201 of the Energy Reorganization Act of 1974 established the NRC as an "independent regulatory commission" and transferred to the NRC "all the licensing and related regulatory functions" of the Atomic Energy Commission, including inspection, enforcement, and the authority to establish safety standards governing the possession and use of radioactive materials. Pursuant to this authority, the NRC independently performs its regulatory mission by issuing regulations, licensing commercial nuclear reactor construction and operation, licensing the possession and use of nuclear materials and wastes, safeguarding nuclear materials and facilities from theft and radiological sabotage, inspecting nuclear facilities, and enforcing regulations and requirements. The NRC also regulates commercial nuclear fuel cycle materials and facilities and licenses 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 Energy Reorganization Act of 1974 transferred all other functions of the Atomic Energy Commission, including its promotional and technology development functions, to ERDA, the predecessor to today's DOE. This division resulted in the complete separation of regulatory responsibilities from promotional responsibilities. The enactment of the Department of Energy Organization Act in 1977 established the DOE by transferring and consolidating several Federal agencies and programs, including ERDA, into a single agency with responsibilities for energy policy, research, and development, including nuclear energy technology and nuclear weapons programs. Over the ensuing decades, the 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 specified in the Energy Reorganization Act, the 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.

The Energy Reorganization Act of 1974 established the NRC as an independent regulatory agency and provided its Commissioners with legal protection from removal. In contrast to the heads of cabinet-level agencies that may be removed from office by the President at will, NRC Commissioners may only be removed for "inefficiency, neglect of duty, or malfeasance in office," not policy disagreements. The NRC has authority to issue its own safety standards governing commercial nuclear facilities and the possession and use of radioactive materials, to conduct its own inspections and oversight of regulated facilities, and to administratively enforce its own regulations without unwarranted influence from others motivated by promotional or economic considerations. For example, as an independent regulatory agency, the NRC is exempt from the interagency regulatory planning and review process within the executive branch, which otherwise requires Federal agencies to submit significant regulatory actions to the OMB for review and clearance before issuance. Additionally, when hearings are held on NRC licensing decisions, agency adjudicators (which can include either the Atomic Safety and Licensing Board or the Commission) are bound to follow strict requirements, such as 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. Furthermore, NRC licensing decisions or final rules governing the conduct of licensees can be challenged in Federal court, where the NRC is entitled to be represented by its own counsel in conjunction with the U.S. Department of Justice. This ensures the agency's interests are always represented when its decisions are challenged by others.

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 to avoid conflicts of interest or creating the appearance of such conflicts. 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 engaged in nuclear fuel cycle activities. 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. Members of the Commission are prohibited by law from engaging in any outside employment during their tenure.

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 13989, "Ethics Commitments by Executive Branch Personnel," dated January 20, 2021, 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 whether 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, investigations, licensee reports, or allegations. 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, dated January 15, 2020. 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:[7]

(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 agency may issue orders for violations or because of a concern involving public health and safety or the common defense and security, or confirmatory orders resulting from alternative dispute resolution.

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.

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

7 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.

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 action is appropriate. 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[8] (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

8 In some cases, it may be appropriate to group violations as examples of a problem in order to appropriately characterize the significance of the event or incident. Hence, this practice informs the licensee and the public that the NRC is aware that the violations are closely related and are not separate regulatory breakdowns.

(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		
	2019	**2020**	**2021**
Notices of violation without civil penalties	11	19	24
Civil penalties	2	7	2
Orders without civil penalties	4	7	0
Total enforcement actions	**17**	**33**	**26**

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.

The 10 CFR 2.206 petition process is a public process, 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 assesses the potential issue and determines whether the requested enforcement action is warranted. 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 300 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 at https://www.nrc.gov/about-nrc/regulatory/allegations-resp.html.

9.4 Openness and Transparency

The NRC established openness as one of five Principles of Good Regulation. The regulatory processes, decisionmaking, and licensee oversight activities are all carried out as transparently as possible. For example, the public, local governmental bodies, Indian Tribes, industry, technical societies, the international community, and citizen groups may participate in a variety of ways before the NRC issues certain licensing actions. To ensure this involvement is meaningful, the NRC meets with the licensees, public and other stakeholders near nuclear facilities, at agency headquarters, and at NRC regional offices. The NRC is also committed to making documents available to the public, unless there is a specific reason for information to be withheld, and to using social media and its Web site extensively to keep the public informed.

NRC requirements are written in a way that allows the agency to carry its day-to-day regulatory oversight and licensing activities in an open and transparent manner. As a result of these objectives, the license holders meet NRC requirements and conduct their activities transparently. Representative examples of regulatory activities that focus on openness, communications, and dissemination of information are discussed below.

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.

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," call for holders of an operating license or combined license for a nuclear power plant to make notifications for various situations that may occur at the site. These reporting requirements include notification of State or local agencies and the public. In 10 CFR 50.72, licensees are required to make reports to the NRC immediately after notifying State or local agencies and not later than 1 hour after the time the licensee declares one of the four emergency classes. Section 16.1.3.1 of this report describes the emergency classifications. In 10 CFR 50.73, licensees are required to submit an LER to the NRC within 60 days after the discovery of an event of the type described in the section. These reports are submitted pursuant to 10 CFR 50.4, "Written Communications," and will be made available to the public unless the content meets the criteria for withholding contained in 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Section 8.1.7 of this report describes the NRC's openness and transparency objectives in more detail.

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. 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. For additional information, see Section 11.2 of this report. 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 2.3.1.8 of this report.

10.2 <u>Probabilistic Risk Assessment Policy</u>

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. The NRC continues to evaluate ways that risk insights can be used to enhance its regulatory framework. Some important elements of this framework include the following:

- The regulations in 10 CFR 50.69 allow licensees to use a risk-informed approach to categorize SSCs and assign special treatment requirements, according to their safety significance. Section 2.3.1.8 of this report discusses this approach in more detail.

- The regulations in 10 CFR 50.48(c) allow an operating nuclear power plant licensee to adopt a risk-informed, performance-based fire protection program. Section 6.3.7 of this report discusses this program in more detail.

- Risk-Informed Technical Specification Initiative 4b enables licensees to make one-time changes to the allowable outage times of safety-related equipment using inputs from PRA models factoring in the real-time status of equipment availability, and Risk-Informed Technical Specification Initiative 5b enables licensees to use inputs from PRA models to modify the surveillance interval of some safety-related equipment using PRA inputs. NUREG-0800, Chapter 16, "Technical Specifications," provides additional detail.

- LIC-504 provides guidance to the staff on how risk information can be used to determine regulatory responses to emerging issues.

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, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, dated December 2020 and the recently issued ASME RA-Sa-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," published in 2021.

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. A Level 1 PRA models various plant and operator responses to initiating events to identify accident sequences that result in reactor core damage, and a Level 2 PRA models and analyzes the progression of severe accidents. 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, as required by 10 CFR Part 54, 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

Level 3 PRA models the release and transport of radioactive material in a severe accident and estimates the health and economic impact in terms of different offsite consequence measures and the associated early and latent fatality risks due to radiation exposure. 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.

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. 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). To enhance the study’s efficiency, the Level 3 PRA project team is leveraging information from approximately the year 2012 on the PWR reference site, the associated reference site’s PRAs, and related research efforts. The study, however, also provides results of a sensitivity study that demonstrates the potential risk reductions resulting from several major modifications (e.g., design and procedural changes to implement FLEX strategies).

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.

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, allows 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. In May 2020, the IAEA published a working document, "A Harmonized Safety Culture Model," that aligned safety culture guidance issued by the NRC, IAEA, World Association of Nuclear Operators (WANO), and INPO.

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 December 14, 2021, NRC inspectors have the option to review licensee self-assessments of safety culture. This inspection procedure 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 September 16, 2020, 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 inspection procedure, 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 March 19, 2021, 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, and its appendix, IP 95003.02, "Guidance for Conducting an Independent NRC Safety Culture Assessment" dated April 1, 2019, which contain requirements and 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 November 17, 2020; 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, which addresses how licensees can consider cybersecurity as part of 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 inspection procedures and performance indicators to ensure that its objectives are being met. The NRC evaluates safety and security interface issues 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 inspection procedures. 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 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.

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 the termination of the NRC-issued license.

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 for nuclear facilities. 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 triggered 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. The next adjustment should take place in 2023.

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). Based on the number of large commercial nuclear power reactors operating as of October 2020, the nuclear power industry is insured to a maximum per-incident dollar level of $13.4 billion under the Price-Anderson framework. As of 2020, the maximum amount of standard retrospective premium for each reactor is $137.609 million per incident (i.e., the $131.056 million maximum deferred premium plus a 5-percent surcharge). With 94 reactor units currently participating in the secondary layer, the total amount of funds available under the secondary layer of financial protection stands at $12.935 billion. The $13.4 billion figure results from adding the maximum amount of available primary insurance of $450 million for the affected site to the maximum available retrospective premium of $12.935 billion. 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. NUREG/CR-7293, “The Price-Anderson Act: 2021 Report to Congress, Public Liability Insurance and Indemnity Requirements for an Evolving Commercial Nuclear Industry,” dated December 16, 2021, contains additional details on the Price-Anderson report.

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.

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 the 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 to maintain a minimum of $1.06 billion in onsite property insurance at each reactor site. 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 qualification, training, and requalification requirements for licensed operators and licensed senior operators under 10 CFR Part 55. The regulations allow 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. 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.

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. Operators and other plant staff also participate in periodic emergency response drills conducted in the simulator and throughout the plant to exercise and evaluate an integrated emergency response-. 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 written examination that covers both the theoretical and site-specific knowledge and abilities required to operate a nuclear power plant. The operating test includes a plant walkthrough and a dynamic performance demonstration on a simulation facility.

In 1999, the NRC amended 10 CFR Part 55 to provide nuclear power reactor licensees the option to prepare the written examinations and operating tests that the agency uses to evaluate the competence of applicants for operators' licenses at those facilities. Most licensees exercise this option. 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 12, dated September 2021. 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 process for nine categories of workers at nuclear power plants, including 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 4, dated June 2019, 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 may 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 licensee would need to report the circumstances of the withdrawal for the staff to determine the significance of the issues related to the withdrawal. 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 qualification and training requirements of 10 CFR Part 50 and 10 CFR Part 55 by (1) inspecting issues at facilities for causes related to training, reviewing LERs, and reviewing 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 2019, 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 ensuring that the people who form the plant staff 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, 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 3, dated July 2020; 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 measurement uncertainty recapture power uprates for Joseph M. Farley Nuclear Plant, Units 1 and 2; Watts Bar Nuclear Plant, Unit 2; Oconee Nuclear Station, Units 1, 2, and 3; and Millstone Power Station, Unit 3. 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 Duane Arnold Energy Center, Pilgrim Nuclear Power Station, Davis-Besse Nuclear Power Station, Perry Nuclear Power Plant, and Beaver Valley Nuclear Power Station.

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. 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.

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 new NRC regulation, 10 CFR 50.155, became effective on September 9, 2019, making the requirements of Order EA-12-049 generically applicable. 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 2.3.3.7 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.

In 2020, the NRC staff approved NuScale's small modular reactor design, which includes control of up to 12 units from a single main control room. The NRC used the guidance in NUREG-1791 to review the results of two separate staffing plan validations. The first staffing plan validation established an initial staffing number. This number was eventually revised by a subsequent topical report that provided evidence supporting safe operation with a reduced number of operators.

The NRC has conducted preapplication activities related to shift staffing with 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 features three crosscutting elements that affect 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.

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, 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. IP 95003 was last updated in 2015.

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 NRC regularly responds to stakeholder and public inquiries and data requests on this system. For example, NRC inspectors have used 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. The NRC 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/. Although new reports have not been added since 2011, the staff maintains the Web site as an historical reference and is working on modernizing the system.

12.2.7 Fitness for Duty

In 10 CFR Part 26, 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:

- Provide reasonable assurance that nuclear power plant personnel are trustworthy and reliable as demonstrated by avoiding substance abuse.
- 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.
- Provide reasonable measures for the early detection of persons who are not fit to perform activities covered by 10 CFR Part 26.
- Provide reasonable assurance that the workplaces are free from the presence and effects of illegal drugs and alcohol.
- 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 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. As a result of the COVID-19 public health emergency, the NRC issued 46 exemptions, at 24 sites, from March 2020 through November 2021. Approved exemptions are posted on the NRC's public Web site at https://www.nrc.gov/about-nrc/covid-19/.

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

In 2021, the NRC issued an amendment to Entergy Operations, Inc., for the Waterford Steam Electric Station, Unit 3, that revised various technical specifications in order for the licensee to implement a planned modification that will replace the digital minicomputers of the core protection calculator system and the control element assembly calculator system with a more reliable digital system. The NRC staff evaluated the human system interfaces as well as the human factors program used to design and evaluate the modification and found them to be consistent with applicable guidance. In 2021, the staff also began preapplication meetings with industry on digital modernization for Turkey Point Nuclear Generating, Units 3 and 4, and Limerick Generating Station, Units 1 and 2. Section 2.3.2.4 of this report provides additional information on these digital upgrades.

12.4.2 Human Performance in Decommissioning Activities

As discussed in Section 2.3.3.4 of this report, on November 3, 2021, the Commission approved publication of a proposed rule on decommissioning activities. The proposed rule was published in the *Federal Register* for public comment on March 3, 2022 (87 FR 12254). The public comment period closes on August 30, 2022. In the proposed rule, 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. 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. Section 2.3.3.4 of this report provides additional information on these rulemaking efforts.

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. Section 13.4 of this report discusses the quality assurance program in more detail.

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. Section 13.3 of this report discusses these companion guides in more detail.

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 <u>Quality Assurance Audits Performed by Licensees</u>

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 October 12, 2021, contains 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 Reactor Oversight

Process monitors reactor licensee performance in three key areas: (1) reactor safety, (2) radiation safety, and (3) safeguards. The Reactor Oversight Process assesses licensee performance using both inspection findings and performance indicators across the seven cornerstones. The NRC determines its regulatory response to licensee performance in accordance with the Reactor Oversight Process Action Matrix using a graded approach that provides for a range of actions commensurate with the safety significance of the inspection findings and performance indicators. The Action Matrix provides 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 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 required 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 processes 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. Other requirements, which may include change control requirements, take precedence over 10 CFR 50.59 for control of specific changes: 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"; 10 CFR Part 50, Appendix B; fire protection license conditions; 10 CFR 50.55a; 10 CFR 50.46; 10 CFR 50.12, "Specific Exemptions"; and 10 CFR Part 20. 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(p). In 10 CFR 50.54(p), the NRC establishes the conditions under which a licensee may make changes to its security plan without prior NRC approval if the changes do not decrease the effectiveness of the plan.

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, Revision 3, endorsed, with clarifications, industry guidance document NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," dated November 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 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 2022, the NRC had approved 171 uprates, resulting in a gain of approximately 24,089 MWt or 8,030 MWe, at existing plants.

Since the issuance of the previous U.S. National Report, the NRC has approved seven measurement uncertainty recapture power uprates totaling 325 MWt (108 MWe) of combined generational capacity for the Joseph M. Farley Nuclear Plant, Units 1 and 2; Watts Bar Nuclear Plant, Unit 2; Oconee Nuclear Station, Units 1, 2, and 3; and Millstone Power Station, Unit 3.

The NRC currently does not have any power uprates under review. In addition, the NRC has not received any indication of anticipated power uprate submittals from licensees within the next 2 years. 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 Rules, 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. Congress set the original 40-year term based on economic and antitrust considerations rather than technical limitations; however, many of the technical safety evaluations were 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 continue to 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, then the schedule for the agency's approval or denial could be affected.

Studies have 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 involved in the requested action. In addition to the review of the renewal application and associated environmental report, the staff performs fact-finding activities through onsite audits and inspections. The NRC documents its safety findings in a safety evaluation report and its environmental findings in a plant-specific environmental impact statement.

Public participation is an important part of the license renewal process. For example, members of the public have opportunities to comment on the staff's draft environmental impact statement. Absent any specific reasons for withholding, information related to the review and approval of a renewal application is publicly available. Any person whose interest might be affected by a license renewal proceeding and who desires to participate as a party must file a written request for hearing and a specification of the contentions (issues) that the person seeks to have litigated. The Commission will grant the hearing request if the Commission finds that the person has standing (that is, is impacted by the license renewal) and has proposed at least one admissible contention.

<u>10 CFR Part 54</u>. The requirements of 10 CFR Part 54 govern the issuance of renewed operating licenses and renewed combined licenses for nuclear power plants. The Commission may issue a renewed license if it finds that the effects of aging will be managed during the period of extended operation and if there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. The Commission must also have assurance that environmental review requirements are satisfied.

The standard for issuance ensures 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 LR-ISG for use by applicants until the guidance is incorporated into the next formal update of the 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, sets the requirements for, among other things, applications for license renewals, and the staff's environmental documents assessing those applications. The environmental review requirements for license renewal under 10 CFR Part 51 are founded on the conclusion that certain environmental issues can be assessed generically and do not need to be reevaluated 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. On a 10-year cycle, the Commission reviews the material in this appendix and updates it if necessary. The NRC publishes the results of its review in the *Federal Register* and invites the public to provide comments and propose other areas that should be updated.

Accordingly, in June 2013, the agency amended 10 CFR Part 51 and its technical basis documented in NUREG-1437, to incorporate lessons learned and knowledge gained from previous license renewal environmental reviews conducted since the NUREG was issued in 1996. The NRC conducts independent reviews of 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 for 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. In February 2022, the Commission directed the NRC staff to update 10 CFR Part 51, NUREG-1437, RG 4.2, and NUREG-1555, as necessary, to include the environmental impacts of renewing the operating license of a nuclear plant for one subsequent license renewal term (i.e., 60-80 years). No subsequent license renewals may be issued without considering those environmental impacts.

The Commission has generically determined that the environmental impacts of continued storage of spent nuclear fuel beyond the licensed life for operation of a reactor are those identified in NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel," dated September 2014. The generic impact determinations for the continued storage of spent fuel in NUREG-2157 shall be deemed incorporated into the supplemental environmental impact statement for license renewal. Further, the supplemental environmental impact statement for license renewal is not required to discuss the need for power or the economic costs and economic benefits of the proposed action or of alternatives to the proposed action except when these benefits and costs are essential for a determination about the inclusion of an alternative or are relevant to mitigation.

14.1.4.2 Experience

The NRC issued the first renewed licenses for the Calvert Cliffs Nuclear Power Plant and the Oconee Nuclear Station in 2000. As of August 2022, 94 reactors have received renewed licenses.[10] As of August 2022, 52 of the 92 operating reactors with renewed licenses that are still currently operating have completed 40 years of operation and are in the period of extended operation. Nine reactor units entered the period of extended operation between 2019 and 2022. Six reactor units are expected to enter the period of extended operation in 2023. 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 https://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.
- 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

10 After receiving their renewed licenses, 10 units have permanently ceased operations: Kewaunee Power Station; Vermont Yankee Nuclear Power Station; Fort Calhoun Station; Oyster Creek Nuclear Generating Station; Pilgrim Nuclear Power Station, Unit 1; Indian Point Nuclear Generating, Units 2 and 3; Three Mile Island Nuclear Station, Unit 1; Duane Arnold Energy Center; and Palisades Nuclear Plant.

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 lessons are learned from the review of renewal applications or generic technical issues are resolved, the NRC issues LR-ISG for use by applicants until the guidance is incorporated into the next formal update of the documents. As discussed in Section 2.3.1.9 of this report, as of November 2021, the NRC has issued four SLR-ISG documents:

- SLR-ISG-2021-01-PWRVI, on reactor vessel internal issues for PWRs
- SLR-ISG-2021-02-MECHANICAL, on mechanical issues
- SLR-ISG-2021-03-STRUCTURES, on structural issues
- SLR-ISG-2021-04-ELECTRICAL, on electrical issues

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. Section 2.3.1.9 of this report describes 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, as described in Section 14.1.5.6.

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, and changes affecting issue finality, 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 whether 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. As discussed in Section 6.3.10 of this report, 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 ensure 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 term of the existing 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 license renewal review focuses on aging management of long-lived, passive structures and components in nuclear power plants (e.g., reactor pressure vessel, steam generators, and piping). The regulation in 10 CFR 50.65 focuses on monitoring the effectiveness of the licensees' maintenance 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 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, and 10 CFR 50.90. These regulations ensure that licensee-initiated changes to the licensing basis are documented and that the licensee obtains NRC review and approval, if necessary. In accordance with 10 CFR 50.59(d)(2), at least every 2 years, the licensee must report to the NRC any changes or modifications to the facility, any changes in procedures, and any changes to tests and experiments made under 10 CFR 50.59(c). 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 annually 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, Forward Fitting, and Issue Finality Processes: 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. MD 8.4 describes the Commission's policies on backfitting, issue finality, and forward fitting. Forward fitting occurs when the NRC conditions its approval of a licensee-initiated request for a licensing action on the licensee's compliance with a new or modified requirement or staff interpretation of a requirement that the licensee did not request.

Backfitting, forward fitting, and changes affecting issue finality are permitted only after a formal, systematic review to ensure that changes are properly justified and suitably defined. The requirements of these processes are intended to ensure order, discipline, and predictability and to optimize the use of NRC staff and licensee resources.

The backfitting, forward fitting, and issue finality processes include an evaluation by the Committee to Review Generic Requirements, 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, forward fitting, and changes affecting issue finality and (2) provide guidance and assistance to the NRC program offices to help them implement the Commission's backfitting, forward fitting, and issue finality policies. Therefore, the review by the Committee to Review

Generic Requirements is a key step in implementing the NRC's backfitting, forward fitting, and issue finality processes, although the primary responsibility for proper backfitting, forward fitting, and issue finality considerations belongs to the NRC staff initiating the backfitting or forward fitting action or change affecting issue finality.

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

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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.

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 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 at https://www.nrc.gov/reactors/operating/oversight.html). 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
- 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's license renewal process and overall regulatory process for nuclear power plants 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, as well as other topics of mutual interest related to materials degradation issues and operating experience. Section 8.1.5.2 of this report provides additional information on the 2010 IRRS mission and the 2014 followup IRRS mission.

Some countries use periodic safety reviews to support the decisionmaking process for long-term operation or license renewal. Section 14.1.5.3 of this report discusses how the NRC uses the Maintenance Rule (10 CFR 50.65) and the License Renewal Rule (10 CFR Part 54) as a robust foundation for this assessment. The NRC participates in key activities to share and obtain international insights relevant to materials degradation and the cumulative effects of plant aging.

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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.

For example, the NRC assessed the results of the first European Union topical peer review, which was carried out from 2017 to 2018 and focused on aging management. Representatives from each participating country provided a national assessment of how aging management programs in their country meet international requirements of aging management. The NRC's assessment found that the United States is generally well aligned with the results of the topical peer review and determined no change in regulatory practices was necessary. The NRC noted that the topical peer review did not address time-limited aging analyses, which are considered of safety importance in the United States.

The NRC has been a key contributor to NEA and IAEA safety standards and guidance documents on aging management. Most relevant has been the staff's involvement in the International Generic Ageing Lessons Learned, which provides a common internationally agreed basis on what constitutes an acceptable aging management program. The NRC's Generic Aging Lessons Learned was the foundation for the International Generic Ageing Lessons Learned, which reinforces that the standards used in the United States conform to those used by the international community. Under the NEA umbrella, the Working Group on Codes and Standards is working toward hosting a workshop on aging management focusing on codes and standards. The NRC's involvement ensures alignment with international practices.

The NRC also actively participates in the IAEA safety review service known as the Safety Aspects of Long Term Operation (SALTO), which comprehensively addresses strategies and technical elements necessary to manage the safety of a nuclear power plant during long-term operation. SALTO missions help countries that operate nuclear power plants ensure that all aspects necessary to manage long-term operation are in place. The NRC shares expertise during these review missions but also benefits by evaluating good practices that may benefit our domestic program. Evaluating SALTO findings in other countries helps ensure that the NRC's existing guidance and regulations are adequate.

Finally, the NRC also recognizes that evaluating and using international operating experience is important to ensure the NRC's regulatory processes conform to international safety practices. The NRC actively participates in meetings of and contributes to the IAEA International Incident Reporting System for Operating Experience and the NEA's Working Group on Operating Experience to gather insights on international safety issues. In addition, the NRC promptly evaluates safety events to ensure plants continue to conform with operating standards. An example would be the actions taken to enhance the safety of nuclear power reactors in the United States following the Fukushima accident in Japan on March 11, 2011. Section 2.3.3.7 of this report gives additional details on the NRC's Fukushima-related accomplishments.

For the reasons summarized above, the United States substantively accomplishes on an ongoing basis the shared objectives associated with periodic safety review and aging management-related guidance from the NEA, IAEA, European Union, European Nuclear Safety Regulators Group, and Western European Nuclear Regulators' Association.

14.2 Verification by Analysis, Surveillance, Testing, and Inspection

Licensees are required to verify that they are operating their nuclear installations in accordance with the plant-specific design and requirements. The technical specifications and national consensus codes (for testing and periodic inspections) contain some of the requirements for verification.

In 10 CFR 50.55a, the NRC 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 Boiler and Pressure Vessel Code and the ASME Operation and Maintenance of Nuclear Power Plants Code. On an approximate 2-year frequency, the NRC updates 10 CFR 50.55a to incorporate by reference the latest editions of ASME Boiler and Pressure Vessel Code Sections III and XI and the ASME Operation and Maintenance 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 2.3.3.7 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 6.3.10 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 37, "Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material"; 10 CFR Part 50 (or comparable 10 CFR Part 52 requirements); 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 regulations in 10 CFR Part 20 establish requirements for radiation protection resulting from activities conducted by 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 establish requirements for 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 needed to maintain adequate protection of public health and safety or provide a cost-beneficial substantial increase in safety or security.

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 sievert (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 37. The regulations in 10 CFR Part 37 establish requirements for the physical protection program for any licensee that possesses an aggregated category 1 or category 2 quantity of radioactive material listed in 10 CFR Part 37, Appendix A, "Category 1 and Category 2 Radioactive Materials." The regulations in 10 CFR Part 37 address background investigations and access authorization for people accessing protected quantities of material and the physical protection of material, including during transit. As described in RIS 2015-15, "Information Regarding a Specific Exemption in the Requirements for the Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material," dated December 4, 2015, licensees with an NRC-approved 10 CFR Part 73 security plan are permitted to rely on the physical protection measures described in that plan to meet the physical protection requirements of 10 CFR Part 37, Subpart B, "Background Investigations and Access Authorization Program," and Subpart C, "Physical Protection Requirements During Use," to the extent that the 10 CFR Part 37 security program provides the equivalent level of protection. In addition, for power reactor licensees, Enforcement Guidance Memorandum 2014-001, "Interim Guidance for Dispositioning 10 CFR Part 37 Violations with Respect to Large Components or Robust Structures Containing Category 1 or Category 2 Quantities of Material at Power Reactor Facilities Licensed under 10 CFR Parts 50 and 52 (RIN 3150-AI12)," dated March 13, 2014, authorizes the NRC staff to exercise enforcement discretion and not cite potential violations associated with protection of material in large components and robust structures if certain criteria are met.

10 CFR Part 50. The regulations in 10 CFR Part 50, such as 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 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," require the NRC to review plant radiation sources, radiation protection design features, and radiation 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 total effective dose equivalent, for evaluating design basis accidents associated with licensing actions that have been submitted to the NRC since 1997. The regulations in 10 CFR 100.11(a) contain 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 approval of packaging and transportation of radioactive material in excess of a Type A quantity and for fissile material. The regulations at 10 CFR 71.5, “Transportation of Licensed Material,” apply the U.S. Department of Transportation’s rules for transportation of radioactive material on NRC licensees. These regulations include 49 CFR Parts 107, 171 through 180, and 390 through 397, as applicable. The current U.S. regulatory framework for the transportation of radioactive material is founded on the standards in the IAEA’s Safety Requirements TS-R-1, “Regulations for the Safe Transport of Radioactive Material,” dated 2009. In 2018, TS-R-1 was superseded by 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 the IAEA’s 2018 edition of the transport regulations in 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 occupationally exposed workers and members of the public. These requirements prescribe exposure limits from radiation, 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, safety, and security 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,”) and exposures to members of the public living near nuclear power plants (NUREG/CR-2907, “Radioactive Effluents from Nuclear Power Plants,”) 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 the Three Mile Island Nuclear Station, 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 0.59 person-Sv (59 person-rem) per reactor, based on a 3-year rolling average basis.

In 2019, 94,237 workers at nuclear plants were monitored for radiation exposure. In 2019, the median collective dose for BWRs and PWRs was 1.01 person-Sv (101 person-rem) and 0.27 person-Sv (27 person-rem), respectively. Of the monitored workers, 38,519 received a measurable dose. The collective measurable does was 50.81 person-Sv (5,081 person-rem) for an average of 0.0013 Sv (0.13 rem) per worker. Of the workers that received a measurable dose in 2019, 84 percent received less than 0.0025 Sv (0.25 rem), 99.9 percent received less than 0.02 Sv (2 rem), and no worker received an excess of 0.03 Sv (3 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 Emergency Plans and Programs

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 accidents can occur and may result in significant offsite radiological release requiring public protective actions. 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 2, dated December 2019, describe the emergency planning basis. The latter guidance document was revised to integrate 35 years of lessons learned into the radiological emergency preparedness program, as well as to consolidate and clarify previous guidance. These criteria provide a basis for licensees and States, Tribal, and local governments to develop radiological emergency plans. 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 emergency response organization staffing assessments. The NRC staff completed the reviews of all licensees' communication assessments by July 2013, and all the staffing assessments by March 2017.

16.1.2 National Response to an Emergency

In May 2013, the response to national emergencies fundamentally changed as a result of DHS's publication of the National Response Framework, the Federal Interagency Operational Plans, and the associated incident annexes, such as the Response Federal Interagency Operational Plan. The DHS revised the operational plans in 2016 and the National Response Framework in 2019. The DHS also 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 may 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 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 agency 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 public health and safety, the NRC has developed a plan and procedures detailing its response to incidents involving licensed facilities, material and activities. 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 may help the State interpret and analyze technical information, update other 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 Incident Response Program at the NRC Headquarters Operations Center near Washington, DC, or the associated regional incident response center, or both. The NRC Incident Response Program team will then (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 Incident Response Program team at the NRC Headquarters Operations Center includes emergency preparedness and response experts and personnel experienced with liaison activities. When the NRC's onsite presence is required, the agency will dispatch Incident Response Program team members, as needed.

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).

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.

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), an applicant or holder of a license for 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 level, 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, is endorsed in RG 1.101, “Emergency Response Planning and Preparedness for Nuclear Power Reactors,” Revision 6, dated June 2021, and provides the latest 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, as endorsed in RG 1.101, is the 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 endorsed in RG 1.101 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.

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 adequate 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 State, Tribal, and local governments to develop acceptable radiological emergency plans.

The NRC and FEMA coordinate their evaluation of periodic emergency response exercises, and the NRC requires all operating nuclear power plant sites to conduct biennial exercises, as discussed in Section 16.1.4 of this report. Under the memorandum of understanding, NRC and FEMA participate in the Steering Committee for Emergency Planning. The steering committee is the focal point for coordination of emergency planning and preparedness, resolving issues between the two agencies, and establishing procedures for assuring the arrangements of the memorandum of understanding are carried out.

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 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. Sections 12.2.3 and 18.1.4, and Part 3 of this report contain additional information on these centers and their equipment.

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 regularly participates in nuclear power plant emergency response and Federal interagency exercises each year to ensure its readiness to respond. The NRC also participates in the planning and conduct of the annual national continuity of operations exercise each year and National Level Exercises on a biennial basis. 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.

The NRC and FEMA coordinate their evaluation of periodic emergency response exercises, and the NRC requires all operating nuclear power plant sites to conduct an exercise on a biennial basis, as outlined in Section IV.F.2 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 in 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.

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 of the Action Matrix. The NRC uses the significance determination process to assess the significance of inspection findings. 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 the baseline and supplemental inspection programs. 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:

- <u>Maintenance of Emergency Preparedness Program</u>—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.

- <u>Drill Evaluation</u>—NRC inspectors evaluate drills and simulator-based training evolutions in which shift operating crews and licensee emergency response organization members participate.

- <u>Exercise Evaluation</u>—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.

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

- <u>Emergency Action Level and Emergency Plan Changes</u>—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.

- <u>Emergency Response Organization Staffing and Augmentation System</u>—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.

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 response 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 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 NRC is arranging with the CNSC to participate in and observe an exercise in Canada in 2022. The "Arrangement between the United States Nuclear Regulatory Commission and the National Nuclear Safety and Safeguards Commission of the United Mexican States 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 will make arrangements with the National Nuclear Safety and Safeguards Commission of the United Mexican States to observe emergency response exercises in the United States and in Mexico when conditions permit.

The NRC routinely practices emergency communications with its Canadian and Mexican counterparts during its emergency drills. The NRC also participates in IAEA Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency Exercises (commonly called Convention Exercises or ConvEx) to test emergency communications and information sharing. In addition, the NRC regularly participates in IAEA emergency preparedness and response conferences, technical meetings, and consultancies. As a result of the pandemic, the NRC was unable to host in-person bilateral exchanges every year about emergency preparedness and response activities and emergency exercise observation with foreign regulatory bodies at NRC Headquarters in Rockville, MD, and at U.S. nuclear power plants around the country. The NRC plans to resume those activities when conditions permit.

The NRC actively communicates with international regulators about emergency preparedness and response for small modular and microreactors, particularly in trilateral discussions with Canada and the United Kingdom. This interaction helps to align on key policy and technical issues. The NRC and its regulatory counterparts share policies, regulatory practices, and experience during these international engagements so the participating countries can gain a common understanding of technical approaches and priorities.

Since 2001, the United States has participated in the International Nuclear and Radiological Event Scale by evaluating operating reactor events and reporting to IAEA any events resulting in a categorization of level 2 or higher. 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. Section 19.6 of this report discusses incident reporting activities and processes.

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, the staff is capturing any changes to NRC review practices resulting from NRC Fukushima lessons learned in guidance updates under already existing processes.

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. 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. The NRC withdrew RS-002, "Processing Applications for Early Site Permits," dated May 3, 2004, because many sections contained outdated guidance that did not reflect the NRC's implementation of a risk-informed, performance-based approach to licensing. After ensuring that all other guidance was reflected in updates to RGs and NUREG-0800, the NRC issued DG-4029, "Use of Plant Parameter Envelope in Early Site Permit Applications," dated June 2021, to provide guidance on the use of the plant parameter envelope concept to postulate certain design parameters for an early site permit application when a specific reactor technology has not been selected for the proposed site.

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; NUREG-0800, Section 3.5.1.6, "Aircraft Hazards," Revision 4, dated March 2010; 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. The onsite meteorological data are used in the estimation of the onsite and offsite atmospheric dispersion values. 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, are used in safety analyses or to establish plant design bases for phenomena such as wind loads or impacts from tornado-generated missiles.

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 10 CFR Part 100 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. In December 2018, the Pacific Earthquake Engineering Research Center issued PEER Report No. 2018/08, "Central and Eastern North America Ground-Motion Characterization—NGA-East Final Report." The new seismic source and ground motion characterization model applied 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. The NRC describes this approach in NUREG-2213, "Updated Implementation Guidelines for SSHAC Hazard Studies," dated October 2018. The updated seismic source and ground motion characterization models provide a consistent and stable basis for developing necessary inputs for probabilistic seismic hazard assessments for the central and eastern United States.

Recent interest in siting nuclear power reactors in regions of the United States with a Quaternary geologic history of volcanic activity led the NRC to publish RG 4.26, "Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites," Revision 0, dated June 2021. RG 4.26 provides applicants with guidance for a risk-informed approach, incorporating guidelines from NUREG-2213, for conducting a probabilistic volcanic hazards assessment that can be used in engineering analysis and siting decisions.

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 February 1983

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

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Plants,” issued June 2003

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

In addition to RGs, 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, APR1400, and NuScale designs with similar results. 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). The NRC considers these impacts through 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.

In accordance with 10 CFR 51.20(b), the approval of a permit or license for the construction or operation of a nuclear power plant, including the site approval process, 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 NRC 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 guidance for the NRC staff to conduct environmental reviews for the applications described above. The NRC is continuing activities to update NUREG-1555, the environmental standard review plan, to align with the updated guidance in RG 4.2, Revision 3. The update will also be conducted for consistency with ongoing rulemaking activities to streamline and enhance the flexibility of the NRC's environmental review process and to document staff generic findings with regard to the construction and operation of advanced nuclear reactors.

The environmental standard review plans in Supplement 1, Revision 1, to NUREG-1555 guide the staff's environmental review for power reactor license renewal applications under 10 CFR Part 54. Article 14 of this report discusses the license renewal process in more detail.

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 six early site permits and eight combined licenses under 10 CFR Part 52.

As part of the revisions to the licensing framework, the NRC issued RS-002 in 2004, which incorporated the environmental guidance in NUREG-1555 and the outcome of interactions with stakeholders. As discussed in Section 17.2.1 of this report, the NRC subsequently withdrew RS-002 as many sections contained outdated language that did not reflect updated NRC licensing approaches. 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. 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 <u>Reevaluation of Site-Related Factors</u>

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 affecting 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. 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 for or tested to the seismic requirements. However, if the equipment has been modified in a manner that likely altered its seismic characteristics, it must be evaluated to determine whether the original seismic qualification still bounds the modified design. IEEE Standard 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," provides the methods for the seismic qualification of equipment to verify the equipment's ability to perform its specified performance requirements criteria to determine the appropriate level of equipment ruggedness. Using this standard, a licensee can determine whether equipment needs to be requalified or replaced.

The NRC published NUREG/KM-0015, "Considerations for Estimating Site-Specific Probable Maximum Precipitation at Nuclear Power Plants in the United States of America," dated September 2021, to summarize the terminologies, theories, general methods, data sources, and procedures used in site-specific probable maximum precipitation development. This document also identifies key considerations in developing and reviewing these estimates that may be used during flooding analyses.

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 reactors. 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 <u>Implementation of Defense in Depth</u>

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
- SRM-SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated July 21, 1993
- SRM-SECY-98-144, “White Paper on Risk-Informed and Performance-Based Regulation,” dated March 1, 1999
- 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/KM-0009, “Historical Review and Observations of Defense-in-Depth,” dated April 2016

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

- RG 1.174, Revision 3

- RG 1.233

- BTP 7-19, Revision 8

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 or human-caused external 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. SRM-SECY-98-144 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
- 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. NRC reviews of these applications ensure that the design of such plants includes design features that provide defense-in-depth. 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, as discussed in Section 13.3.3 of this report. To confirm compliance with quality assurance and defect reporting requirements, the NRC interacts with manufacturers and suppliers of safety-related components through its vendor inspection program and oversees nuclear power plant construction. The NRC has developed an inspection program for nuclear plants that incorporates ITAAC, along with lessons learned from the inspection program used in the previous construction era (1970-1980) and from the construction of Watts Bar Nuclear Plant, Unit 2. Inspection Manual Chapter 2503, “Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work,” dated October 6, 2020, describes these inspections. The NRC staff will verify successful ITAAC completion based on these inspections.

The use of an adequate vendor inspection program and an adequate quality assurance program, are three examples of defense-in-depth applied to the early stages of plant construction. Upon completion of construction, successful completion of ITAAC assures that the defense-in-depth measures designed into the facility have been correctly constructed in accordance with the design.

The NRC has been using these defense-in-depth principles during its construction oversight of two AP1000 units, Vogtle Electric Generating Plant, Units 3 and 4, which are located in Waynesboro, GA and owned by Southern Nuclear. The combined license enables the licensee to construct a plant and to operate it if certain conditions, or ITAAC, are satisfied.

Originally, each combined license for a Westinghouse AP1000 design contained approximately 875 ITAAC. Overall, ITAAC consolidation licensing activities have reduced the number of ITAAC in the combined licenses for the Vogtle Electric Generating Plant, Units 3 and 4, by approximately 55 percent.

Southern Nuclear notified the NRC that all ITAAC for the Vogtle Electric Generating Plant, Unit 3, were complete, as required by 10 CFR 52.99(c)(4), on July 29, 2022. In accordance with 10 CFR 52.103(g), the NRC found that the ITAAC for the Unit 3 combined license were met on August 3, 2022. Accordingly, Southern Nuclear may now operate the Vogtle Electric Generating Plant, Unit 3, including loading fuel in accordance with the terms of the license. The NRC staff transitioned its oversight of Unit 3 from the Construction Reactor Oversight Process to the Reactor Oversight Process after it made the 10 CFR 52.103(g) finding.

As of Juluy 2022, Southern Nuclear has completed approximately 31 percent of the ITAAC for Vogtle Electric Generating Plant, Unit 4. The licensee must complete all ITAAC included in the combined license and notify the NRC staff before the NRC can make the finding required by 10 CFR 52.103(g) for that unit.

As discussed in Section 2.3.3.3 of this report, in March 2018, the NRC formed the Vogtle Readiness Group to identify and resolve any licensing, inspection, or regulatory challenges or gaps that could impact the schedule for completion of the construction project. The Vogtle Readiness Group brings together different NRC organizations that carry separate responsibilities related to the regulation of the project. It provides high-level assessments, coordination, oversight, and management direction of NRC activities associated with the licensing, inspection, testing, and operation of the new units.

Separately, in October 2019, the NRC established the Vogtle Project Office in the Office of Nuclear Reactor Regulation. The Vogtle Project Office is directly responsible for licensing and project management for the construction and startup of the Vogtle Electric Generating Plant, Units 3 and 4, and it coordinates closely with the NRC's Division of Construction Oversight on implementation of the construction inspection and oversight programs. The Vogtle Project Office staff developed the steps and guidance to support the staff's ability to make an effective and timely finding under 10 CFR 52.103(g). In preparation for the first-ever finding, the staff focused on assessing program readiness and monitoring the effectiveness of previous program changes. The staff in the Vogtle Project Office and Division of Construction Oversight also conducted training for inspectors to ensure that any technical issues identified late in the construction schedule were prioritized to ensure timely resolution. In addition, the staff held tabletop exercises to work through scenarios for the effective and timely handling of late construction or ITAAC inspection findings and late-filed allegations.

The Vogtle Project Office also exercised routine communications with Southern Nuclear to maintain readiness to review licensing actions that the licensee might need to support construction. For example, by the time the NRC issued the 10 CFR 52.103(g) finding for the Vogtle Electric Generating Plant, Unit 3, the agency had reviewed and approved more than 185 license amendments, along with a number of exemption requests and alternatives to ASME Code requirements. In addition, to promote public engagement, the staff conducts routine public meetings, typically weekly, to discuss licensing activities and issues related to ITAAC for the construction project. For public transparency, the staff also continues to provide monthly resource expenditure reports on the NRC's public Web site and has completed enhancements to its Web sites to improve clarity and make navigation easier.

To process the late construction surge of Unit 3 ITAAC closure notifications required by 10 CFR 52.99, both the Vogtle Project Office and the Division of Construction Oversight employed a number of strategies to ensure timely review of these notifications. For example, the staff established an expanded pool of notification reviewers to handle the expected increase toward the end of construction. The staff also routinely engages with Southern Nuclear on the schedule for ITAAC completion to ensure inspectors are readily available to inspect key ITAAC activities as they are conducted.

Due to restrictions resulting from the Federal, State, and local response to the COVID-19 public health emergency, the staff from the Division of Construction Inspection adapted the inspection program to ensure its effectiveness while protecting the health and safety of inspectors during the public health emergency, including through the extensive use of remote inspections. Inspectors have since returned to conducting onsite inspections of ITAAC, the initial test program, and operational programs based on safety significance and the uniqueness or complexity of the construction activity.

Finally, the staff has implemented a lessons-learned initiative to conduct a holistic assessment of the 10 CFR Part 52 licensing and construction oversight and inspection programs for the purpose of improving the effectiveness and efficiency of future programs. This initiative focuses on evaluating the NRC's construction inspection, ITAAC, and licensing effectiveness, highlighting actions that contributed to the success of construction program implementation and identifying areas where improvements can be made.

18.1.4 Experience and Implementation of Defense-in-Depth Measures

The NRC has long recognized the importance of the defense-in-depth philosophy and has implemented regulations to establish and strengthen defense in depth in the nuclear industry. In an operating facility, compliance with technical specifications and control of the information in the FSAR information assures that defense-in-depth is maintained. An event such as the loss of electrical power, as occurred at Fukushima Dai-ichi, can be an important contributor to the risk of power plant accidents. This risk was identified in NUREG-75/014 and addressed in 10 CFR 50.63, "Loss of All Alternating Current Power" (the Station Blackout Rule), in 1988. The conditions and duration of blackout assumed in 10 CFR 50.63 were proven to be insufficient for an event similar to the Fukushima Dai-ichi accident; however, this rule was a first step in beyond-design-basis accident mitigation.

Following the terrorist events of September 11, 2001, the NRC updated its regulations (10 CFR 50.54(hh)(2), now 10 CFR 50.155(b)(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. Additionally, 10 CFR 50.150, "Aircraft Impact Assessment," introduced the requirement for licensees and applicants with licenses or construction permits dated after July 13, 2009, to assess the impact of a large commercial aircraft to demonstrate core cooling, containment, and SFP integrity are maintained.

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. Following publication of the Near-Term Task Force report, the NRC issued three orders on March 12, 2012, two of which were codified in 10 CFR 50.155. Order EA-12-049 required a three-phased approach for mitigating beyond-design-basis external events to maintain or restore key safety functions, and Order EA-12-051 imposed design features and requirements for reliable SFP level instrumentation. Additionally, the NRC issued requests for information on March 12, 2012, 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 U.S. nuclear industry proposed the FLEX initiative to develop an integrated safety-focused approach to expedite implementation of Fukushima lessons learned. 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 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 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 from off site for a longer term) in responding to an accident and ensuring equipment availability and redundancy. As part of the initiative, the industry 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.

Through the rulemaking process, the NRC developed and implemented 10 CFR 50.155, which made the requirements in Orders EA-12-049 and EA-12-051 generally applicable and addressed issues raised by petitions for rulemaking that were submitted to the NRC because of the Fukushima Dai-ichi accident. Of note, the rule does not mandate the three-phase approach required by Order EA-12-049. The rule requires each applicant or licensee to develop, implement, and maintain mitigation strategies for beyond-design-basis external events and extensive damage mitigation guidelines—specifically, guidelines to mitigate external events from natural phenomena assuming a loss of all AC power concurrent with loss of normal access to the ultimate heat sink, or normal heat sink for passive reactor designs, and strategies to restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant impacted by the event. The rule also establishes requirements for equipment, training, and SFP monitoring to ensure sufficient response to such events. The NRC withdrew Orders EA-12-049 and EA-12-051 with the promulgation of 10 CFR 50.155.

Section 2.3.3.7 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.

As with large light-water reactors, the NuScale small modular reactor design has a leak-tight containment, which houses the reactor and steam generator, all of which are part of a “module.” The purpose of the containment is to protect against uncontrolled releases of radioactivity to the environment and ensure that any leakage from the reactor to the outside environment is controlled subsequent to a design-basis accident so that the containment does not exceed the allowable leakage rate given in the technical specifications. Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to 10 CFR Part 50, governs containment integrity and testing. This regulation specifies that containments must be tested before initial reactor operation and periodically tested thereafter to verify their leak-tight integrity. The testing includes pressurizing the containment to peak accident pressure and measuring the leakage rate (Type A testing). Also, local leakage rate testing (Types B and C) of containment isolation valves and flange seals is conducted at peak accident pressure, which does not require pressurizing the containment vessel. Since this testing is done during refueling outages, Type A testing significantly extends the duration of the outage since it cannot be done until all refueling activities are complete and the module is ready to be moved from the refueling area to its operating location in the reactor building.

The NuScale containment vessel is proposed to be fabricated, and pressure tested at the factory, as an ASME Code Class MC (metal containment). Allowable leakage is zero. This allows the vessel to be certified as an ASME Code Class 1 leak-tight pressure vessel. As a certified ASME Code vessel, as long as periodic inservice inspections are conducted, the containment remains certified as a leak-tight (zero-leakage) vessel. The containment is 100-percent inspectable, both inside and outside. Therefore, periodic Type A containment testing is not required, reducing refueling outage time. Based on the fabrication methodology and continued inservice inspections, NuScale requested and received an exemption from Appendix J to 10 CFR Part 50.

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 3, 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.1.2 of RG 1.206 addresses the human factors engineering review of combined license applications.

18.3.2 Experience

In August 2020, the NRC approved NuScale Power LLC’s 12-module small modular reactor design certification. As part of NuScale’s efforts to update and evolve its design, NuScale submitted licensing Topical Report (TR)-0420-69456, “NuScale Control Room Staffing Plan,” Revision 0, dated June 11, 2020; and Revision 1, dated December 17, 2020. In this topical report, NuScale requested that the staff approve a control room staffing plan with a minimum control room crew of three licensed operators and no shift technical advisor. The NRC staff evaluated NuScale’s proposed topical report and finalized its technical review and associated final safety evaluation report in May 2021. The NRC staff found that TR-0420-69456, Revision 1, is acceptable for referencing in licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the final safety evaluation.

18.3.2.1 Human Factors Engineering

The NRC’s human factors engineering reviews for design certification applications focus on evaluating either 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 or reports that summarize the results of human factors engineering activities. When implementation plans are submitted, 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 https://www.nrc.gov/reading-rm/doc-collections/isg/digital-instrumentation-ctrl.html.

The staff has completed its safety reviews of the instrumentation and control systems for the AP1000, ESBWR, and APR1400 reactor designs as well as those for the Fermi, Unit 3 combined license. As requested by the applicant, the staff suspended the review of the U.S. APWR design certification. The staff also completed reviewing the instrumentation and control design for the ABWR design certification renewal. The staff continues to support the instrumentation and control ITAAC inspection activities for the AP1000 combined licenses at the Vogtle Electric Generating Plant, Units 3 and 4. The staff also continues to review digital instrumentation and control-platform topical reports that can be referenced in subsequent design certification and 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. The design-specific review standard chapter on instrumentation and controls reflects some important lessons the staff learned when using NUREG-0800 to review new large light-water reactor instrumentation and control designs. In addition, this guidance emphasizes fundamental instrumentation and control design principles of independence, redundancy, predictability and repeatability, and diversity and defense in depth.

The staff developed and successfully used the design-specific review standard for evaluating the NuScale design certification application. 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 staff completed its final safety evaluation report, and the NRC approved the NuScale design in 2020. The design is currently undergoing the final certification through rulemaking. Successful use of the design-specific review standard and the lessons learned in evaluating the NuScale instrumentation and control design formed a significant basis for developing the design review guide highlighted in Section 2.3.2.4 of this report.

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 participates in NEA the Working Group on Digital Instrumentation and Control, which promotes harmonization and improvements in nuclear safety through the development of regulatory guidance for instrumentation and control topics and technical issues of concern to its member countries, for both operating and new reactors.

Section 2.3.2.4 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 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 2015, the NRC and the North American Electric Reliability Corporation renewed a memorandum of understanding to address nuclear plant cybersecurity roles, responsibilities, and areas of coordination between the two organizations. The memorandum of understanding discusses how 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 and have developed the next version with updates to information sharing, emergency response, and security. The agreement is expected to be renewed in the summer 2022.

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 completed the inspections in 2021. The NRC developed IP 71130.10, "Cyber Security," dated

January 1, 2022, to assess ongoing performance of the cybersecurity program. Based on lessons learned from the last 11 years of operating experience and the results of a 2019 self-assessment of the cybersecurity program implementation, the NRC, with input from the NEI, is updating the available guidance to improve the regulatory efficiency and effectiveness of cybersecurity at U.S. nuclear power plants.

The NRC is working with national laboratories to conduct research projects to further improve the regulatory efficiency and effectiveness of cybersecurity oversight. These research efforts will help the NRC address cybersecurity requirements for advanced reactors through the development of a new risk-informed and technology-inclusive regulatory framework. Specifically, the NRC is currently exploring a performance-based, graded approach through a new rule and regulatory guidance development that could accommodate the wide range of advanced reactor technologies.

The NRC interfaces with other Federal agencies and with State agencies to increase awareness of the cyber-threat landscape of the Nation. These agencies include the DHS's Cybersecurity and Infrastructure Security Agency, the Federal Energy Regulatory Commission, NIST, U.S. intelligence and law enforcement communities, and other agencies involved with cybersecurity issues.

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 have been 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, has 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. 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 six early site permits and two limited work authorizations, which allow the permit holder to perform limited construction activities at a site. The staff has approved one early site permit for the Clinch River Nuclear Site since the issuance of the last U.S. National Report.

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. In August 2020, the NRC approved NuScale Power LLC’s 12-module small modular reactor design, which is now undergoing certification through a rulemaking process. Review of Mitsubishi’s U.S. APWR application has been suspended. In March 2020, the NRC staff completed the technical review of General Electric’s ABWR design certification renewal application. As discussed in Section 6.3.1 of this report, the ABWR design certification renewal became effective in September 2021.

Additionally, on February 27, 2021, the AP1000 design certification in Appendix D to 10 CFR Part 52 expired. As discussed in Section 6.3.1 of this report, the staff proposed, and the Commission approved, an extension to this certification. With this extension, the AP1000 design certification remains valid for referencing until February 27, 2026.

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. The NRC currently has no combined license applications 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 analyses and evaluation in the safety analysis report and amendments submitted. 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. 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 nuclear steam supply system vendor-specific standard technical specifications in NUREG-1430 through NUREG-1434, Volumes 1 and 2, dated September 2021, 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, a majority of the operating commercial nuclear plants has 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. 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. 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. Each nuclear facility is required to follow the quality assurance requirements in Appendix B to 10 CFR Part 50, and many licensees' technical specifications require the licensee to establish, implement, and maintain procedures consistent with RG 1.33 or the approved quality assurance topical report, which typically conforms with ANSI 3.2-2012 and ANSI N18.7-1976.

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."

The nuclear industry also developed severe accident management guidelines (SAMGs) in response to the Three Mile Island Nuclear Station 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. 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, emergency classification, and emergency action levels.

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.

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 SAT-derived training programs for instrumentation and control, electrical maintenance and mechanical maintenance personnel, including engineering support personnel. The NRC verifies the adequacy of these programs before 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 and 10 CFR 50.73 help to achieve these goals, as 10 CFR 50.72 requires immediate notification requirements through the emergency notification system, and 10 CFR 50.73 requires 60-day written 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 staff uses the information that is required to be reported by 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 share operating experience with the industry. Evaluations of events as documented in NRC inspection reports are available on the NRC's public 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 report required 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 helped to focus 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 2021, only two U.S. reactor events were significant from the standpoint of public health and safety, as defined by the abnormal occurrence criteria in NUREG-0090: "Report to Congress on Abnormal Occurrences:"

- On June 7, 2011, at Fort Calhoun Station, an improperly replaced electrical breaker resulted in a fire that 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, so the program description here is also broadly applicable to the review of construction experience.

As discussed in Section 2.3.2.4 of this report, the NRC is exploring ways to improve the data collection process using Mission Analytics Portal-External as a portal for licensee submission of notifications. In addition, the Data Warehouse provides an onsite repository of data that can be more easily transformed and extracted to allow connections between datasets that had previously been siloed. This has provided a substantial improvement in the structured data readily available for analysis and communication.

The NRC facilitates the collection, storage, and retrieval of operating experience data through an internal Web site containing the Operating Experience Hub, which provides a centralized repository of links to databases, dashboards, and analytics relevant to operating experience. These links provide search features for event reports, inspection findings, analytics for risk significant events, human factors events, and reactor scrams, as well as general search functions for operating experience communications. Since 2010, the NRC has collected additional information in a broader database that provides the same type of centralized data storage and retrieval options for nonreportable 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. INPO’s Industry Reporting and Information 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. In addition, these events are included in dashboards and other analytics tools available on the Operating Experience Hub for easy access and retrieval during consideration of similar issues. 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 inspection procedures.

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. 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 Radioactive Waste

The U.S. Government addresses in detail the spent fuel and radioactive waste programs, including high-level waste, in a report prepared to satisfy the reporting requirements of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The latest report, "United States of America National Report for the Seventh Review Meeting of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management" (Joint Convention National Report), dated October 2020, is available at https://www.energy.gov/sites/default/files/2020/10/f80/7th-JC-RM-United-States-NR-Final-Oct-2020.pdf.

The NRC has issued regulations and guidance for nuclear power reactor licensees to ensure the safe management and disposal of low-level radioactive waste. The term "radioactive waste" includes radioactive waste in liquid effluents, gaseous effluents, and solid waste.

Applications for Licenses—Design of Radioactive Waste Facilities. As discussed in Section 15.2 of this report, licensees are required to limit effluent releases in accordance with regulations such as 10 CFR 50.34a, 10 CFR 50.36a, and through technical specifications requirements in Appendix I of 10 CFR Part 50.

NUREG-0800, Chapter 11, “Radioactive Waste Management,” provides information on the design of radwaste systems and guidance related to solid waste form, characterization, and classification. Onsite waste storage facilities should be sized to provide sufficient storage capacity and allow sufficient time for the decay of shorter lived radionuclides before shipping in accordance with the following guidance:

- NUREG-0800, BTP 11-3, “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants,” Revision 4, dated January 2016

- RIS 2004-17, “Revised Decay-in-Storage Provisions for the Storage of Radioactive Waste Containing Byproduct Material,” Revision 1, dated September 27, 2005

- RIS 2008-32, “Interim Low Level Radioactive Waste Storage at Reactor Sites,” dated December 30, 2008

- RIS 2011-09, “Available Resources Associated with Extended Storage of Low-Level Waste,” dated August 16, 2011.

RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 2, dated November 2001, provides guidance on methods acceptable to the NRC for the design, construction, installation, and testing of SSCs of radioactive waste management facilities. In addition, RG 4.21 contains guidance on submitting design information and operational procedures for minimizing radioactive waste generation and contamination of the facility and the environment, and facilitating decommissioning.

<u>Radioactive Material Effluent Controls, Reporting, and Procedures</u>. Licensees are required by 10 CFR 50.36a, plant technical specifications, and license conditions to keep average annual releases of radioactive material in liquid and gaseous effluents and resultant doses at small percentages of the public dose limits. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the Commission may require the licensee to take action as the Commission deems appropriate to reduce radioactive effluent releases. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to ensure that the public is provided a dependable source of power, even under unusual conditions that may temporarily result in releases higher than such small percentages, but still within the public dose limits.

NRC guidance on the measuring and reporting of liquid and gaseous radioactive waste is given in RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Revision 3, dated September 2021. The following documents provide the NRC’s guidance on measuring radioactive material in the environment:

- RG 4.1, “Radiological Environmental Monitoring for Nuclear Power Plants,” Revision 2, dated June 2009

- RG 4.13, “Environmental Dosimetry – Performance Specifications, Testing, and Data Analysis,” Revision 2, dated June 2019

- RG 4.14, “Radiological Effluent and Environmental Monitoring at Uranium Mills,” Revision 1, dated April 1980

As discussed in Section 15.3 of this report, the regulations in 10 CFR 50.36a require licensees to annually report to the NRC the quantity of principal radionuclides released to the unrestricted area in liquid and gaseous effluents and to estimate the maximum potential annual radiation doses to the public annually. The data from these effluent reports is summarized annually in NUREG/CR-2907. The most recent effluent and environmental monitoring report for each nuclear power plant is provided on the NRC Web site at https://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

RG 1.33 discusses the quality assurance program requirements, which include procedures for operation of the liquid and gaseous radioactive waste system, and the solid waste systems. The NRC also requires licensees to have a process control program that contains the sampling, analysis, and formulation for the solidification of radioactive waste from the liquid radioactive waste systems.

As discussed in Section 6.3.8 of this report, the NRC Decommissioning Planning Rule and 10 CFR 20.1406 require licensees to conduct their operations in a way that minimizes the introduction of residual radioactivity into the site, which includes the site’s subsurface soil and groundwater. The Decommissioning Planning Rule also updated 10 CFR 20.1501 to require licensees to perform radiological surveys, including the subsurface (e.g., groundwater). The rule recognizes that relatively large volumes of low specific activity may need to be stored and disposed.

Solid Radioactive Waste Generation and Onsite Storage. On May 1, 2012, NRC 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.

As discussed in Section 6.3.8 of this report, under the Decommissioning Planning Rule and RG 4.22, licensees of operating facilities are required to minimize contamination and radioactive waste generation, conduct appropriate radiological surveys including of the subsurface, maintain records of residual radioactivity, and provide adequate funding to complete decommissioning. The NRC last updated the low-level waste storage guidelines in RIS 2011-09.

Solid Radioactive Waste Shipments. Waste containers, shipping casks, and methods of packaging wastes are required to meet all applicable Federal regulations, which include the following:

- 10 CFR 20.2006, “Transfer for Disposal and Manifests,” and to 10 CFR Part 20, Appendix G, “Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests,” addressing the transfer and manifesting of radioactive waste shipments

- 10 CFR Part 71, addressing the packaging and transportation of radioactive materials

- 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste,” or in corresponding State regulations, addressing applicable waste acceptance criteria of the disposal facility or waste processors

- 49 CFR Parts 171 through 180, addressing U.S. Department of Transportation regulations for the shipment of radioactive materials

NRC Inspection Program. As discussed in Section 15.2 of this report, the NRC conducts inspections under the public and occupational radiation safety cornerstones to ensure that requirements are being met.

Offsite Waste Disposal. 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. 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, 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.

Spent Nuclear Fuel. 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. Consolidated interim storage facilities are facilities proposed for the interim storage of spent fuel and reactor-related greater than Class C low-level radioactive waste before final disposal in a deep geologic disposal facility. The consolidated interim storage facilities would be similar to existing independent spent fuel storage installations providing dry storage of spent fuel with integrated shielding structures. Consolidated interim storage facilities will be regulated under 10 CFR Part 72, and, as proposed, would not be co-located with a power reactor. In April 2016, Interim Storage Partners, LLC, submitted an application to the NRC for a specific license to construct and operate a consolidated interim storage facility at a site in Andrews, TX. In March 2017, Holtec International submitted an application to the NRC for a specific license to construct and operate the HI-STORE consolidated interim storage facility, to be located in Lea County, NM. In September 2021, the NRC issued a license to Interim Storage Partners, LLC, to receive, possess, transfer, and store up to 5,000 metric tons of uranium of spent fuel and 231.3 metric tons of greater than Class C low-level radioactive waste for 40 years. The HI-STORE license application is still under NRC review, and the NRC has not yet made a licensing decision. The NRC has posted additional information on consolidated interim storage facilities on its Web site at https://www.nrc.gov/waste/spent-fuel-storage/cis.html.

The U.S. currently has no facility for spent fuel or high-level waste disposal. In 2008, the DOE applied to the NRC for authorization to construct a geologic repository at Yucca Mountain, NV, for spent fuel and high-level waste disposal. On March 3, 2010, the DOE filed a motion to withdraw its license application, which the Atomic Safety and Licensing Board denied on

13 The NRC’s 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 the DOE in a yet-to-be-determined manner. Until then, such waste must be managed (stored) by licensees. Regulations in 10 CFR Part 72 address the onsite management of greater than Class C low-level waste in independent storage facilities.

June 29, 2010. The Commission was evenly divided on whether to uphold or overturn the Board's decision and took no affirmative action. However, in recognition of budgetary limitations, the Commission directed the Board to complete all necessary and appropriate case management activities, and the Atomic Safety and Licensing Board suspended the proceeding on September 30, 2011. The NRC resumed work on its technical and environmental reviews of the Yucca Mountain application using available funds in response to an August 2013 ruling by the U.S. Court of Appeals for the District of Columbia Circuit. In January 2015, the NRC staff completed its safety evaluation report, which is documented in five volumes of NUREG-1949, "Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada":

(1) NUREG-1949, Volume 1, "General Information," dated August 2010

(2) NUREG-1949, Volume 2, "Repository Safety Before Permanent Closure" dated January 2015

(3) NUREG-1949, Volume 3, "Repository Safety After Permanent Closure" dated October 2014,

(4) NUREG-1949, Volume 4, "Administrative and Programmatic Requirements" dated December 2014

(5) NUREG-1949, Volume 5, "Proposed Conditions on the Construction Authorization and Probable Subjects of License Specifications," dated January 2015

The staff also completed and issued NUREG-2184, "Supplement to the U.S. Department of Energy's Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada," dated May 2016.

The safety evaluation report includes the staff's recommendation that the Commission should not authorize construction of the repository because the DOE has not met certain land and water rights requirements identified in NUREG-1949, Volume 4. Completion of the safety evaluation report does not represent an agency decision on whether to authorize construction. Currently, the adjudication remains suspended. Should additional funds be appropriated and the adjudication resumed in the future, pending contentions challenging the DOE's application would need to be resolved, and the Commission would need to complete its review before reaching a final licensing decision.

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 2022

INPO®

1 Executive Summary

The U.S. nuclear power industry established the Institute of Nuclear Power Operations (INPO or "the Institute") in 1979 to promote the highest levels of safety and reliability—*to promote excellence*—in plant operation following a fuel-damaging event at Three Mile Island Nuclear Generating Station. The Institute is a nongovernmental corporation that operates on a not-for-profit basis. Under U.S. tax law, the company is classified as a charitable organization that "relieves the burden of the Government."

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 21 members. In addition, many utility organizations that jointly own U.S. nuclear power plants are associate members. Several 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: it assumed the function of overseeing INPO activities while endowing INPO with ample authority to shape industrywide performance. This feature makes INPO unique. The Institute accomplishes its mission in four ways.

(1) It establishes industry "excellence standards" by developing performance objectives and criteria (POs&Cs) and Tier 1 INPO Event Reports (IERs) and Principles documents.

(2) It measures and compares industry performance against those standards.

(3) It assists in industry improvement initiatives.

(4) It exercises authority over its members when it must. The industry's recognition that all nuclear utilities are affected by the action of any one utility has motivated its continuing support of INPO.

The U.S. Nuclear Regulatory Commission (NRC) has statutory responsibility for overseeing 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. Both organizations consider each individual member solely responsible for the safe operation of its nuclear plants.

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 2021, the U.S. nuclear industry was performing at its highest levels ever. Today, the median industry capability factor is above 93 percent; most plants experience no automatic scrams in a year (2021 was the lowest year on record for scrams); and collective radiation dose and industrial accident rates are both steady, near all-time lows. Going forward, INPO has implemented a new 10-year strategy aimed at sustaining this high performance while also promoting continual improvement and increasing additional safety and reliability margins.

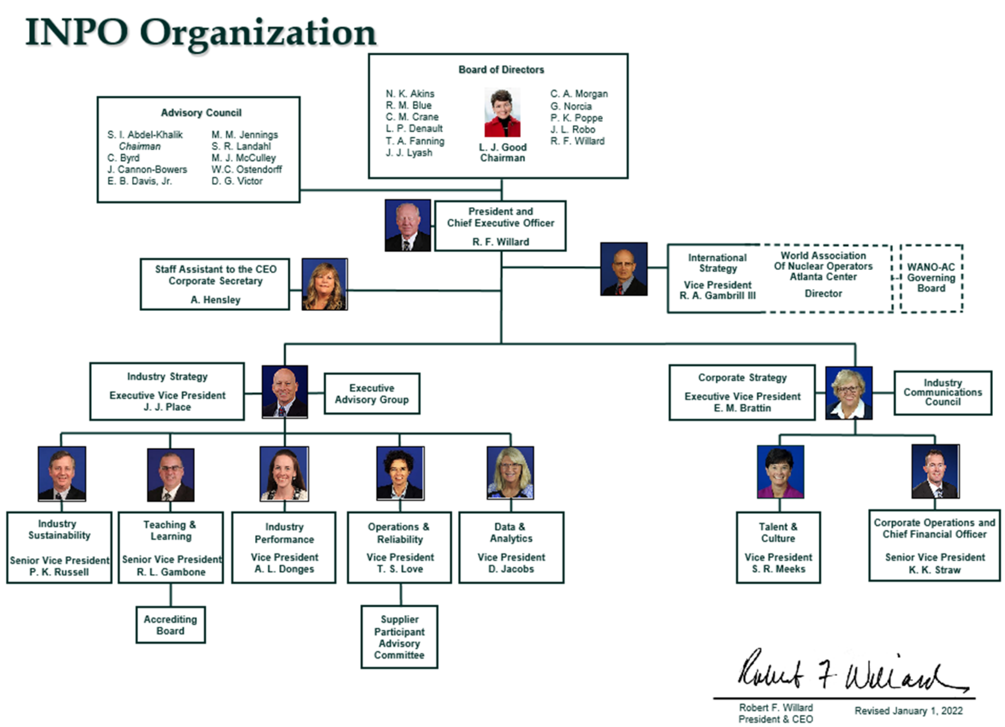
Despite record performance levels, challenges persist that warrant more attention from the industry and INPO. For example, while the number of lower performing plants in the industry has significantly decreased, several persist, and others have not yet fully recovered to top-level performance. Also, while the industry has made progress in reducing the number of consequential events attributable to equipment, operations, maintenance, and engineering, more work is needed to ensure sustainable plant reliability across the board. Similarly, while significant progress was made to minimize the number of plants operating with fuel defects (2021 was a record year), additional effort will be needed to sustain this performance. The Institute continues working closely with industry stakeholders to close these remaining performance gaps.

Finally, the industry has taken numerous actions 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 Daiichi nuclear plant. The Institute conducts periodic reviews to ensure that U.S. industry actions are sustained and to evaluate each member station's ability to respond to extreme external events.

2 Organization, Governance, and Strategy

2.1 <u>Organization and Governance</u>

In many ways, INPO's organizational structure resembles that of a typical U.S. corporation. A board of directors, comprised of the INPO chief executive officer (CEO) and 13 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. The chart below depicts INPO's organizational structure.



Because the INPO board consists utility executives, the industry believes that it is important for the board to have support from an advisory council of diverse individuals from outside the nuclear industry. The INPO Advisory Council of 9 to 15 professionals meets periodically to review the Institute's activities and to advise both the INPO management staff and the Board of Directors. Advisory Council members include prominent educators, scientists, engineers, business executives, and experts in organizational effectiveness, human relations, and finance.

The Institute ensures that the industry actively participates in its programs and initiatives. Representatives from member utilities serve on an INPO's Executive Advisory Group and Industry Communications Council. The Executive Advisory Group, comprised of chief nuclear officers from all the member organizations, advises INPO management personnel in nuclear technical areas and on INPO's operations. The Industry Communications Council advises on the effectiveness of communication of INPO programs and activities. The Institute also operates the National Academy for Nuclear Training. The Academy Council provides advice in the areas of training, accreditation and human performance. INPO frequently establishes ad hoc industry groups to provide input on specific technical issues or improvement initiatives.

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

(1) *CEO engagement*: A fundamental element in the creation of INPO was the personal involvement and support of member CEOs. Today, the same level of CEO support and involvement remains essential to INPO's continued influence on the industry.

(2) *Nuclear safety*: The Institute's mission to promote the highest levels of safety and reliability—to promote excellence—in the operation of commercial nuclear power plants has not wavered. A focus on nuclear safety is at the forefront of every INPO activity. The distinction between excellence and mere regulatory compliance informs the industry's continuous improvements in nuclear safety and reliability.

(3) *Broad industry support*: The nuclear industry was involved in developing standards of excellence and is committed to meeting those standards. Each member accepts that, as part of the self-regulation model, its nuclear stations are subject to onsite evaluations involving industry peers. The evaluations are intrusive, comprehensive and performance based. The industry also supports and participates in self-regulation through continuous performance monitoring; involvement in advisory groups, industry task forces, and working groups; and by the loan of employees to INPO. This gives participants firsthand experience and knowledge about improvement opportunities at their own sites, while increasing their understanding of INPO's role and the importance of self-regulation across the industry.

(4) *Accountability*: Institute evaluations, assessments and continuous monitoring lead to an understanding of industrywide performance that, in turn, prompts peer-to-peer accountability and the identification of plants and corporate organizations that require special assistance to improve their performance. INPO evaluation results also affect member insurance rates.

(5) *Independence*: Although it is part of the U.S. nuclear power industry, INPO remains independent in its evaluative and shaping activities. The Institute establishes high

industry standards and distinguishes clearly between its self-regulatory role in enforcing those standards and its other collaborative interactions and activities with its members.

(6) *Confidentiality*: The Institute and its member utilities recognize that it is essential for them to maintain an environment that allows for critical peer reviews leading to self-improvement. Frank communications with utility staff members, which are central to the evaluation and monitoring processes, depend on the assurance that the information will be used privately and constructively. Misuse of information in INPO reports by individuals outside of the industry would be detrimental to INPO's ability to obtain information from its members and to candidly identify needed improvements.

2.2 INPO's Strategy

The Institute is committed to a long-term strategic design that outlines how it will fulfill its mission through 2030. The strategic design considers the current states of the INPO corporate organization and of the U.S. and international industries, the desired end states in 2030, and potential barriers to achieving the desired outcomes in each of the separate, but interrelated, areas. Within the strategic design are priorities and measurable outcomes that guide the application of INPO's limited resources.

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

The Institute's corporate responsibilities encompass the strategic bases for shaping U.S. and international industry performance—together with the traditional corporate tasks of equipping the INPO workforce with the knowledge, skills and resources to execute its mission. INPO's corporate strategy includes the following elements:

- The Institute is committed to developing and maintaining a diverse workforce and attracting top-performing employees whose talents match Institute and industry needs. INPO's effectiveness is largely determined by the quality of its people and their engagement with coworkers, members and other external stakeholders; as such, INPO must maintain a workforce that is recognized as excellent in today's environment.

- Understanding that a strong culture has a powerful influence on performance, INPO strives to instill a culture that emphasizes integrity, accountability, and high performance. The Institute's core values focus on behaviors that not only enable its mission but also establish a culture of inclusion—for all its benefits—and of deep commitment to continuous improvement.

- The Institute employs a technology infrastructure commensurate with its mission and strategy requirements. This infrastructure remains effective and maintainable through comprehensive needs analyses, careful planning and execution, and regular investment. As INPO's strategy and operating model evolve, its technology infrastructure must be continually adapted and strengthened to support its mission needs.

2.4 Financial and Human Resources

INPO's operating budget of $107 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.

The Institute's permanent staff of about 305 full-time employees is augmented extensively by industry professionals who serve as loaned employees or as international liaison engineers on assignments of 18 to 24 months. Loaned and liaison employees comprise about one-third of the total technical staff. By working at INPO, they gain extensive experience and training while providing current industry expertise and diversity of thought and practice. A small number of permanent INPO employees serve on loaned assignments to member organizations, primarily for professional development. The total number of permanent and loaned employees at INPO is approximately 375.

The Institute's resources and capabilities are further extended by U.S. and international utility peers and executive industry advisers who participate in a wide range of short-term activities, including performance monitoring, evaluation and accreditation visits to nuclear plants. These peers offer INPO teams varied perspectives informed by their current utility or plant experience. In turn, they learn how other U.S. industry stations perform and how those organizations approach problems. In 2022, the industry will provide INPO with an estimated 650 peers for short-term assignments.

2.5 INPO's U.S. Industry Strategy

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

The Institute's industry-facing strategy currently addresses continuous improvement and sustainability challenges in four areas:

(1) *Performance outliers*: Fundamental to a high-performing nuclear industry is the capability to identify and eliminate performance shortfalls that contribute to consequential events and raise risk. Nuclear leaders must address performance shortfalls, whether they are broad-based or isolated to specific units or functional areas. Chronic low or outlier performance must be corrected. The Institute and the industry, including suppliers, must identify low performance and poor quality wherever it exists and eliminate it.

(2) *Sustainable industry performance*: Underpinning a belief in *continuous improvement in pursuit of performance excellence* are five core values that are described in INPO 19-003, "Staying on Top—Advancing a Culture of Continuous Improvement," issued August 2019. While all five core values contribute to sustainable performance, a focus on four of them—*self-awareness/self-correction, excellence standards, talent development,* and *continuous learning*—will achieve the greatest gains.

(3) *Data science and analytics*: Data science is in increasing use across the nuclear industry, but with limited effect as yet. Data-driven insights from state-of-the-art analysis, modeling and visualization are needed to support decisions that will improve industry performance and focus resources. Governance and oversight must be improved to enable consistency and reduce the risks of unauthorized disclosures and data corruption. Finally, if the nuclear workforce has the competencies to maximize the use of the available tools, it will operate more effectively and efficiently and with minimum risk.

(4) *Teaching and learning*: Today, nuclear industry education imparts the basic knowledge needed for a qualified, effective workforce. Industry training in simulators, in laboratories, and on the job reinforces the skills needed to perform high-quality work. To date, this approach has contributed to record high U.S. industry performance. However, to sustain these gains in the long term and to further advance safety and reliability, the National Academy for Nuclear Training must begin to graduate a cadre of motivated leaders whose passion for teaching and learning will elevate and sustain industry proficiency and performance to a level of *excellence* characterized by the highest standards, a culture of continuous learning, and rapidly advancing improvement initiatives across their companies and stations.

2.6 International Strategy

INPO leverages and supports the World Association of Nuclear Operators (WANO) to promote improved nuclear safety worldwide and to allow U.S. operators to benefit from worldwide operating experience. The Institute operates WANO-Atlanta Centre (WANO-AC) and conducts all operations and activities to meet or exceed WANO program requirements.

The Institute works to reduce global nuclear industry risks by promoting WANO's programs and by exemplifying the attainment of WANO's strategic goals at WANO-AC. In addition to operating WANO-AC, INPO provides direct and indirect support to the London office and other WANO regions.

The WANO Action for Excellence initiative highlights the need for WANO to fully develop capabilities to recover low-performing stations, diagnose leadership and organizational gaps, conduct continuous performance monitoring, and advance leadership and talent development. The Institute is actively sharing techniques for developing these capabilities.

3 INPO's Role within the Federal Regulatory Framework

The Federal Government regulates nuclear utilities in the United States as it does other industries that could affect the health and safety of the public. For the nuclear industry, 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 event. The Kemeny Commission, as it came to be known, helped to influence the industry's decision to create INPO as a means of self-regulation. In the broadest sense, the NRC and INPO have the ultimate goals, as they both strive to protect the public by promoting safe and reliable plant operations. Consequently, some important areas of nuclear plant performance are reviewed by both organizations. However, INPO in no way supplants the regulatory role of the NRC.

From its inception, INPO recognized that it would need to work closely with the NRC without becoming or appearing to become an extension of, or adviser to, the Government. In recognition of their differing roles but common objectives, the NRC and INPO established a memorandum of agreement that included plans for coordinating areas of mutual interest so as not to confuse or complicate matters for the industry or either party. The industry also acknowledged the need for the NRC to assess the quality of INPO products and programs. Consequently, the memorandum of agreement contains provisions for the following:

- copies of select generic documents to be exchanged
- access to common data, such as specific elements of INPO's Industry Reporting and Information System (IRIS)
- observation of certain INPO field activities (such as evaluations) by NRC employees, upon agreement from the affected members
- observation of National Nuclear Accrediting Board sessions

The Institute regularly participates in industry-led working groups and task forces that interface with the NRC on specific regulatory issues and initiatives relevant 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 focus on their respective activities. For example, the NRC's Reactor Oversight Process use operating data collected by IRIS, which is run by INPO. This lets the NRC avoid redundant data collection.

Although the NRC inspection regime may appear to overlap with INPO plant evaluations and WANO peer reviews, these activities serve different purposes. The NRC inspects for compliance with Federal standards in very specific performance areas, while INPO's evaluative activities assess performance more broadly against standards of excellence that often exceed Federal guidelines. These differences are discussed in annual engagements between between NRC executives and members of INPO leadership.

Finally, INPO has implemented a policy and procedures for handling matters that are reportable to the NRC. The Institute alerts utility management personnel of any such issues so that utility organizations can evaluate and report the issues themselves. However, if INPO becomes aware of a failure to comply with Federal regulations, it assumes an obligation to ensure that the issue is reported to the NRC if the utility organization has not already reported it.

4 Responsibilities of INPO and Its Members

Members of INPO 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 establishing and maintaining accredited training programs for personnel who operate, maintain and support their nuclear plants. Members are expected to address all "areas for improvement" identified through INPO evaluation, accreditation, continuous performance monitoring, and operating experience programs.

Nuclear operators are explicitly responsible for complying with the terms and conditions of their operating licenses and the applicable rules and regulations. Each licensee is ultimately responsible for the safety of its activities and the safeguarding of nuclear facilities and materials used in operations. These regulatory tenets remain foundational to INPO's relationship with its members.

A specific INPO policy outlines actions to be taken if a member is unresponsive to INPO, is unwilling or unable to take action to resolve a significant safety issue, or has persistent shortfalls in performance, or if the accreditation for its training programs has been placed on probation or withdrawn by the National Nuclear Accrediting Board. The policy specifies that INPO and the member utility's management team should work together to resolve these issues, using a graduated approach of increasing accountability. Specific options 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 needed, suspension would significantly affect the utility's continued operation of its nuclear assets, 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. For example, they should provide INPO with detailed and timely operating experience information and should participate fully in the loaned employee, peer evaluator and WANO programs.

In return, the industry expects INPO to provide members with results from evaluations, continuous performance monitoring, and accreditation and review visits, including written reports and an overall numerical assessment of performance relative to standards of excellence. The industry expects INPO to follow up on members' corrective actions and to verify that these have been implemented.

The Institute 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 outside their organizations. The Institute also expects members and participants to use information from INPO to improve nuclear operations, not for other purposes (e.g., for commercial advantage). Members are to avoid including INPO or INPO documents in litigation.

Members of INPO that also belong to the collective insurance organization Nuclear Electric Insurance Limited (NEIL) have authorized and instructed the Institute to make available to NEIL copies of its evaluation reports and other data, which NEIL reviews for issues that could affect its members' insurability.

The INPO POs&Cs are written with industry input and support but without regard to utility-specific constraints or agreements, such as labor agreements. The Institute expects each member to resolve any impediments that outside organizations may impose on the implementation of the POs&Cs.

The Institute does not engage in public, media or legislative activities to promote nuclear power, as such activities could appear to undermine INPO's objectivity and credibility and could 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 electricity 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 emulation, continuous improvement, and the sharing of best practices.

The Institute facilitates information-sharing by including industry peers in nearly all of its programs, including plant evaluations, training and accreditation, continuous performance monitoring, and plant recovery. The Institute shares information through various channels, including the secure 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 types of information are private and not appropriate to share. Examples are plant-specific details of INPO evaluation and accreditation results, personal employee and individual performance information, and marketing data.

The INPO international strategy calls for it to provide all INPO Principles and Guideline documents and IERs (which are discussed later) to WANO. The extent of sharing has been curtailed recently because of increasingly complex U.S. export control regulations.

6 Priority to Safety Culture

The U.S. nuclear industry is deeply committed to the tenets of nuclear safety culture in recognition of the special and unique nature of nuclear technology and its associated hazards: radioactive byproducts, concentration of energy in the reactor core, and decay heat. At INPO, the behaviors that are essential to a strong safety culture have been embedded in everything the Institute has done 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 December 2012, INPO distributed INPO 12-012, "Traits of a Healthy Nuclear Safety Culture," developed in collaboration with WANO and the NRC. This report superseded "Principles for a Strong Nuclear Safety Culture," issued in November 2004.

In April 2013, INPO developed and distributed two addenda to INPO 12-012. Addendum I, titled "Behaviors and Actions That Support a Healthy Nuclear Safety Culture," lists behaviors and examples found in INPO 12 012, but sorted by organizational level and attribute. Addendum II, titled "Cross References for Traits of a Healthy Nuclear Safety Culture," cross references the traits identified in INPO 12 012 with NRC safety culture components and International Atomic Energy Agency safety culture characteristics.

The industry uses INPO evaluations and other activities to identify and help correct early signs of decline in the safety culture of any plant or utility. The industry has defined INPO's role as follows:

- Define and publish standards relative to safety culture.
- Evaluate safety culture at each plant or utility.
- 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 annual INPO CEO Conference.

Safety culture is thoroughly examined during INPO's evaluation and continuous monitoring processes. The Institute expects each INPO evaluation team to review the safety culture of the plant or utility throughout the evaluation process, including during the preevaluation analysis of plant data and observations. The results of this review are included in the summary of organizational effectiveness. The evaluation team discusses aspects of a safety culture with the CEO of the utility at each evaluation exit briefing. Similarly, monitoring leaders review safety culture during their continuous monitoring activities. The results of this review are included in the INPO Performance Summary Report, and are discussed with station leaders.

7 Operations, Activities, and Actions

In the execution of its strategic design, INPO conducts a 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.

7.1 <u>Evaluation Programs</u>

Members host regular INPO plant evaluations or WANO peer reviews of their nuclear plants approximately every 2 years. Teams from INPO also periodically conduct review visits on other, more specific areas of plant operations. During these evaluations and reviews, the INPO teams compare performance with excellence standards—such as the POs&Cs—their own experience, and their broad knowledge of industry best practices. The standards of excellence guide the evaluation processes and are the bases for identified areas for improvement.

7.1.1 Plant Evaluations

Historically, teams of approximately 18 to 25 qualified and experienced individuals conduct evaluations of operating nuclear plants.

The scope of an 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 are not exclusive to functional area 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)
- operating experience
- organizational effectiveness (leadership, team effectiveness, management)

Lastly, teams evaluate the following foundational areas:

- nuclear safety culture
- nuclear professionals
- leadership fundamentals

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 leadership and management personnel using a model that defines organizational effectiveness (leadership, teamwork, and management), nuclear safety culture, technical conscience, and nuclear oversight topics.

A key part of each evaluation includes observing the performance of operations and training personnel during simulated exercises. In addition, evaluations include—where practicable—observations of refueling outages, plant startups and shutdowns, major planned evolutions, and planned fire and emergency preparedness drills.

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

The size of INPO's performance-based evaluation teams is determined by station performance. A station with exemplary performance may be evaluated by a base team consisting of six individuals, while 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 (simulated) abnormal and emergency conditions.
- Preevaluation observations are conducted during outages or at other times when station workload is higher.
- WANO program requirements are met.
- 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 2021, 13 performance-based evaluations were conducted. In 2022, of the 32 evaluations and peer reviews scheduled, 18 will be WANO peer reviews, and 14 will be conducted using the performance-based approach.

After each evaluation, the evaluation team continues to provide the utility with formal reports of strengths and areas for improvement, and INPO continues to provide numerical assessments. Assessments range 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. INPO provides these reports and assessments because utility CEOs and managers want to know precisely how their station's performance compares to the standards of excellence. The process is consistent with INPO's responsibility to its members and CEOs to identify low-performing nuclear plants and to stimulate continuous performance improvement.

In addition to providing a summary of station performance, since 2020, evaluation teams provide a summary of the ability of the station to sustain its performance. Using INPO 19-003 as a standard, the evaluation team evaluates the strength of the continuous improvement culture present at the station by analyzing its performance over time and by observing current behaviors. The conclusions of the evaluation team are included in the formal report and are briefed to the utility CEO and station leaders.

The final report includes the utility's responses to the identified areas for improvement and their commitment to specific corrective actions. In subsequent evaluations and other interactions, INPO specifically reviews the effectiveness of the utility's past actions to correct performance.

The Institute's department managers also provide area performance summaries that assess current performance, any contrasting areas of performance, and the utility's trajectory or anticipated near-term future performance. The trajectory assessment forecasts whether performance will improve, remain stable, or decline in coming months, based on organizational effectiveness, proficiency of leaders and workers, and workload.

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.

In the past decade, U.S. industry performance has risen to historically high levels. Numerous improvements have been made in plant safety and reliability by addressing issues identified during evaluations, peer reviews, and plant self-assessments, and through 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 in the lower assessment categories has substantially declined.

Prestartup reviews are conducted at each new unit before initial core fuel loading and initial criticality of the reactor. The purposes of a prestartup review are as follows:

- to determine whether a new unit is ready to start safely and reliably
- to characterize the initial performance of previously started new units at the same site during prestartup reviews of subsequent units
- to recharacterize the readiness of a new unit when the prestartup or startup process was delayed for more than 1 year
- to verify that the operating organization and, in particular, the operators have moved from a construction mindset to one that makes nuclear safety the overriding priority

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 whether plant personnel are ready to safely shut down the plant and remove the nuclear fuel to its interim storage location..

7.1.2 Corporate Evaluations

INPO recognizes that the corporate office and both nuclear and nonnuclear corporate leaders have a strong influence on safe and reliable nuclear operation. The Institute conducts corporate evaluations at 6-year intervals, with a follow-up performance review 2.5 to 3 years after each corporate evaluation to verify progress on identified weaknesses.

A tailored set of POs&Cs and a Tier 1 Principles document on excellence in corporate performance define the standards for assessing corporate performance. 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
- strategic direction that defines the utility organization's expectations for station operations—including business and operational plans—and performance standards
- corporate support for major plant modifications
- integrated risk management
- corporate and independent oversight of station performance
- performance of corporate functions, such as human resources, industrial relations, fuel management, supply chain management, and other areas applicable to the nuclear organization

The Institute's 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.

Between corporate evaluations, INPO oversees nuclear corporate organizations through continuous monitoring (discussed later). Where appropriate to improve performance, INPO also provides assistance, including benchmarking of the corporate activities of other high-performing members.

Upon invitation from its members, INPO meets with utility boards of directors to provide an overview of plant and fleet performance. The boards compare these briefings with views from independent oversight groups and corporate inputs to complete their assessment of nuclear performance.

7.1.3 Other Review Visits

The industry also leverages INPO to conduct technical review visits in select industrywide problem areas. Teams are led by INPO and often include industry subject-matter experts. Review visits typically include a week of preparation followed by a week on site. Examples of areas reviewed include materials issues that may affect the structural integrity of the reactor coolant system and reactor vessel internals, and components or systems that are significant contributors to unplanned plant transients and forced outages. Review visits have sometimes led to detailed technical guidance for each utility to implement.

Review visit reports often identify beneficial practices and make recommendations for improvement. These reports are sent to station site vice presidents; for safety-significant recommendations, INPO may request a response. Subsequent plant evaluations or WANO peer review teams follow up on each recommendation requiring a response to ensure that identified issues are appropriately addressed.

Periodically, INPO posts the beneficial practices and recommendations on the secure member Web site to allow all utilities a benchmarking opportunity.

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. The Institute initiated review visits to target sites that were contributing to the adverse trend. The purpose of these review visits was to observe operators in training and on the job to determine what was driving weaknesses in operator fundamentals. INPO completed more than 20 review visits in 2017 and 2018. Thanks to the review visits and a related IER, 2018 saw fewer operator fundamental events and sustained industrywide improvement.

Pressurized-Water Reactor Materials Review Visits: The Institute began conducting review visits targeting the steam generator as early as 1996. Throughout the 1980s, steam generator tube leaks and ruptures contributed to lost generation and were the cause of several events deemed significant by INPO. Through the Electric Power Research Institute's (EPRI's) Steam Generator Management Program, the industry issued detailed guidance on qualifications for and implementation of nondestructive testing techniques, engineering assessments of steam generator integrity, and detection of and response to tube leakage and ruptures. In mid-1995, the industry requested that INPO assist in improving prevention and detection of steam generator degradation by ensuring more consistent implementation of industry guidance and by evaluating steam generator management. As a result, INPO established the Steam Generator Review Visit Program.

The EPRI Pressurized-Water Reactor (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. In 2003, INPO launched the Primary System Integrity Review Visit Program in response to several notable events associated with leakage from PWR borated systems resulting in corrosion and wastage of pressure-barrier components in the reactor coolant system.

In 2012, INPO combined the Steam Generator Review Visit Program and the Primary System Integrity Review Visit Program into the PWR Materials Review Visit Program to capture all aspects of the industry initiative codified in Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues," Revision 3, issued February 2017. This initiative encompasses the Steam Generator Review Visit Program, the EPRI PWR Materials Reliability Program, and other programs directly dealing with primary system materials. While the review visit scope and team size are larger, the objective remains the same: ensuring nuclear safety and plant reliability are not compromised by weaknesses associated in the primary pressure boundary, including the steam generators.

In 2016, the scope of the PWR Materials Review Visit Program was expanded to take an even broader look at materials degradation, including flow-accelerated corrosion and buried pipe and tank integrity.

Boiling-Water Reactor Materials Review Visits. In 2001, INPO initiated the Boiling-Water Reactor (BWR) Vessel and Internals Review Visit Program 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 after the establishment of these review visits, as evidenced by the lack of safety-significant events and events contributing to lost generation.

In 2016, the scope of the BWR Vessel and Internals Review Visit Program was expanded to take an even broader look at materials degradation, including flow-accelerated corrosion and buried pipe and tank integrity. To reflect this scope change, the name of the program is being changed to the BWR Materials Review Visit Program.

In 2018, INPO piloted a centralized materials review visit at the corporate office, consolidating the reviews of 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.

<u>Alternating Current Power Source Reliability Review Visits</u>. 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 alternating current (AC) power reliability. Three to five loss of offsite power matrix reviews are 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 loss of offsite power 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.

In addition, an improving trend has emerged in fewer full and partial loss of offsite power 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 loss of offsite power 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.

The Institute 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. The Institute is also engaged with EPRI in the industry's 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. Main Generator 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 during plant evaluations. Teams focused on performance and condition monitoring to ensure that the generator was operating within design parameters and that monitoring was in place to detect early signs of equipment degradation. Institute personnel remain engaged in industry working groups and in emergent plant issues related to main generator, turbine, and support systems.

Fuel Integrity Review Visits. The Institute performed Fuel Integrity Review Visits in 2017 and 2018 to gather detailed information regarding fuel integrity performance in the U.S. fleet. Specific sites that had experienced recent fuel failures were chosen 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 of the station's fuel rod failures and the corrective actions being taken by each station organization. Recommendations and beneficial practices are identified and documented in a report that is issued to the station. INPO personnel also followup on the station organization's response to these recommendations.

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 the 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.

7.2 Continuous Monitoring

In the second half of 2014, INPO established a performance monitoring program that uses all available data in combination with targeted, systematic engagement and 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 helps 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 with no training program accreditation probations.

A team of performance monitoring leaders continuously reviews and analyzes stations' performance data 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. All performance monitoring leaders collaboratively review performance twice per quarter. Additionally, INPO senior leaders review select stations once per week. Each performance monitoring leader is responsible for monitoring approximately six stations that are grouped by fleet organization. When signs of decline are identified, the performance monitoring leader works with station leaders and INPO leaders to develop a plan to arrest the decline and improve performance.

INPO expanded the Continuous Performance Monitoring Program in 2018 to include corporate performance, and again in 2019 for WANO-AC stations outside the United States.

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

(1) *Monitor*: Monitoring leaders use all available data and information to characterize station and corporate performance. Integrating data with plant observations and with insights from other touchpoints allows the performance monitoring leader to develop a comprehensive picture of station performance. Credible trigger points are used to identify developing gaps that require attention. Station leaders receive an INPO Performance Summary Report (IPSR) twice each quarter, WANO-AC stations outside the United States receive an updated IPSR once per quarter, and corporate leaders receive a Corporate IPSR once per quarter. The IPSR summarizes the current integrated picture of station performance from INPO's perspective.

(2) *Engage*: Monitoring leaders engage station leaders, primarily site vice presidents, to exchange views on performance issues and the effectiveness of corrective actions.

(3) *Intervene*: When called for, intervention may be required to shape performance improvement. In the case of a precipitous decline, the plant may be assigned to INPO's plant performance recovery organization. Performance recovery uses additional tools and techniques that rely more on direct observations of station performance and on more interactions with station leaders.

In 2020, based on a recognition that industry performance had significantly improved and on feedback from the industry, INPO launched an effort to innovate the continuous monitoring process. The new model for continuous monitoring was designed to provide a deeper, more comprehensive picture of station and corporate performance and sustainability to focus monitoring and assistance efforts on the organizations that are at greatest risk of decline. The associated changes were aimed at making INPO teams more scalable and agile in meeting the needs of the industry and at reducing the burden on high-performing, sustainable stations that consistently demonstrate a culture of continuous improvement.

As part of this effort, INPO developed a heat map tool that plots station or corporate performance versus sustainability. This provides greater insight into the risk a station or corporation poses to the industry. Sustainability is determined by assessing the values outlined in INPO 19-003. Performance is determined using several inputs, including Neural Plant Performance Indicator models, scrams, consequential events, and fuel performance.

In 2021, INPO revised Policy Note 14, "INPO/WANO-AC Engagement," which covers engagement across the full spectrum of industry performance and sustainability. The policy note outlines a graduated approach to engagement, in which lower performing stations and corporate organizations receive increased engagement, and higher performers are expected to demonstrate increased self-reliance. Additionally, the policy note includes descriptions of updated engagement categories—Monitoring, Augmented Monitoring, and Full Monitoring—which are defined as follows:

- *Monitoring*: For stations and corporations characterized by exemplary performance and behaviors reflecting a culture of continuous improvement. They pose the lowest risk to the industry and consistently demonstrate the organizational capability and capacity to identify and correct performance weaknesses. As such, INPO and WANO-AC interactions with these organizations are routine.

- *Augmented Monitoring*: For stations and corporations that demonstrate exemplary or strong performance and behaviors that largely reflect a culture of improvement but that nonetheless pose some increased risk to the industry. Performance or sustainability gaps exist that require specific increased engagement and that warrant the station organization or corporation to implement a detailed improvement plan. In these cases, the Institute and WANO-AC will monitor progress through routine interactions. When progress is inadequate or performance or sustainability gaps are substantial, INPO and WANO-AC will establish a targeted assistance plan to accelerate improvement.

- *Full Monitoring*: For stations and corporations that pose a higher risk to the industry have wider or deeper performance or sustainability gaps, and generally do not exhibit the behaviors of a continuous improvement culture. They generally lack the capability or capacity to improve performance and/or sustainability without a structured plan that includes INPO and industry assistance. In specific cases, a station may be placed in Special Focus, a subset of Full Monitoring.

In 2020, INPO began revising its operating model to reflect the U.S. industry's current high performance and to incorporate agile organization and engagement methods that could be scaled up or down as the industry changed. The redesigned operating model reduced INPO's industry-facing operations from six large-scale operations to four, including the introduction of a new operation entitled the Performance Continuum. The Performance Continuum strategically integrates evaluations and peer reviews, continuous monitoring, and training and accreditation into one overall operation. The Performance Continuum shifts INPO away from being primarily time-based, whereby operations are regularly scheduled and executed in a one-size-fits-all manner, to an approach where activities are tailored to stations and corporate organizations based on their performance and sustainability.

In addition to strengthening collaboration within INPO, the Performance Continuum will promote higher levels of collaboration within the industry. Central to the continuum is the opportunity for INPO to take the wealth of data that it has as well as the advancements it has made in analytics and visualization to the next level—applying a higher level of advanced analytics to enable the work INPO does.

7.2.1 Performance and Sustainability

Commencing in 2020, performance monitoring leaders began developing perspectives on sustainability for each station and corporation to include a narrative summary and associated numerical value. Using INPO 19-003 as a standard, performance monitoring leaders review the station's or corporation's culture of continuous improvement by analyzing its performance over time and observing current behaviors. The performance over time is used as an initial indicator of sustainability, confirmed by an assessment of behaviors. Inputs that provide insights on behaviors include observations, both on-site and remote, and interviews.

7.3 Pandemic Impact—INPO and Industry

7.3.1 INPO Response to the Pandemic

At the onset of the Coronavirus Disease 2019 (COVID-19) pandemic, INPO employees were instructed to work from home, and department managers established regular virtual touchpoints with employees. The Institute staffed a pandemic response center, consisting of three positions, which operated Monday through Friday and published daily situation reports. INPO also established a pandemic response team to monitor the pandemic and to provide policy recommendations to INPO's leaders.

From the onset, priority was given to the industry's lower-performing stations, and INPO teams continued to engage both virtually and in person when travel was not restricted. INPO leaders also directed that all scheduled events that could be executed, either traditionally or creatively, should be, so as not to create a bow wave of unfulfilled commitments. Well-developed operations like continuous monitoring enabled INPO to remain informed of industry performance despite the pandemic.

Some plant evaluations, peer reviews, and continuous monitoring visits that were scheduled between March and July 2020 were rescheduled. Quickly, however, innovative methods were deployed to observe plant and corporate performance using remote means. These included conducting INPO leadership seminars with larger industry audiences virtually. During the second half of 2020, in-person INPO interactions increased using a graded approach based on plant and corporate performance. By August 2020, travel activities were largely restored, with some restrictions. International travel recommenced in October 2020.

In November 2021, the INPO staff returned to in-office work for most days of the week, and INPO's protective measures were adjusted based on COVID-19 conditions and on the assessed effectiveness of mitigating actions. By December 2021, most deferred INPO evaluations and peer reviews had been completed. Lessons learned were evaluated, and useful remote interactions with plants and corporations were adopted for use in INPO's new operating model. Many of these practices are expected to remain in effect post-pandemic.

7.3.2 Industry Response to the Pandemic

Beginning in March 2020, domestic member stations began various pandemic responses, such as implementing infection control measures, canceling non-essential business (for example, site tours and visits), and increasing pandemic-related communications. Some outage work was deferred in the spring and fall 2020 outages based on availability of supplemental workers. Most of the deferred work involved optional activities used to balance outage scope such that the activities returned to their original schedules.

In the early phases of the pandemic, most U.S. INPO members and WANO-AC members outside of the United States began implementing stay-at-home and remote work measures for nonessential personnel. Foreign travel became restrictive. At many sites, access to control rooms was limited to essential workers, and other mitigating measures were implemented that altered some work practices. Temperature monitoring stations and self-screening protocols were widely established to minimize the risk of introducing sick workers to sites. Most INPO activities were regarded as essential, so field activities at U.S. stations were minimally affected by the added restrictions.

7.3.3 Effects on Industry Performance

At the onset of the pandemic (March-April 2020), INPO identified an increasing trend of preventable events—some consequential—including scrams. A lack of focus by workers and less oversight of risk-significant activities were determined to be contributors.

The Institute communicated with industry executives and shared the range of preventable events involving operations and maintenance personnel. Key communications included the following:

- Self-awareness by workers of their state of readiness to perform their assigned tasks must be assessed. In other words, their proficiency must be understood and accounted for.

- Managers and supervisors must know the proficiency of their teams and must take all necessary steps to mitigate any potential shortfalls.

Industry actions were effective in arresting the trend, and higher performance was recorded during the second half of 2020 and throughout 2021.

The Institute participated in weekly chief nuclear officer communications to stay apprised of the range of actions and issues confronting industry leaders. The Institute also closely monitored work deferrals to determine whether these were creating a bow wave of new work that the industry would be challenged to overcome. The Institute concluded that work deferrals were being adequately managed.

Despite the challenges posed by the pandemic, the U.S. nuclear industry achieved its highest ever level of performance during this period, as indicated by objective measures, including a record low number of scrams in 2021 and a reduction in the number of consequential events.

7.4 Training and Accreditation Programs

The U.S. commercial nuclear power industry is strongly committed to the education and training of plant operators, maintenance workers and other groups of workers. For this reason, in 1985 the industry established the National Academy for Nuclear Training (hereafter “the Academy”), which operates within INPO. The focus of the Academy is to promote professionalism of nuclear plant personnel and to ensure that education and training achieve the highest possible standards of nuclear qualifications and job performance.

The Academy integrates the training-related activities of all members, including the accreditation of specific training programs that are overseen by an independent National Nuclear Accrediting Board. Each U.S. utility becomes a member of the Academy when all its operating plants achieve accreditations for all applicable training programs. Through INPO, the Academy also conducts seminars and courses and provides other education and training materials for use by member utility personnel.

The Institute conducts evaluations of accredited training programs, performs activities to verify that standards of accreditation are maintained, and provides assistance at the request of member utilities. Written objectives and criteria are jointly developed with the industry and guide the accreditation process. The independent National Nuclear Accrediting Board periodically examines the quality of utility training programs and makes all decisions regarding accreditation. If training programs meet accreditation standards, the National Nuclear Accrediting Board awards or renews accreditation. If the program has significant problems, the National Nuclear Accrediting Board may defer initial accreditation, may place the program on probation, or may withdraw accreditation. Accreditation is maintained on an ongoing basis and is formally renewed for each of the applicable training program every six years.

Although the NRC remains independent of INPO and the accreditation process, it acknowledges the value of the National Nuclear Accrediting Board in upholding standards that meet the requirements of the NRC's Training Rule. To assist in maintaining its alignment with regulatory guidelines, the Accrediting Board enlists the support of ex-NRC executives as participants, permits the NRC to observe Boards, and—upon request—joins INPO teams that evaluate accredited training programs in the field.

7.4.1 Courses and Seminars

The industry benefits extensively from courses and seminars that the Academy conducts. A tiered set of courses and seminars is provided to the industry to develop leadership skills, for personnel from first-line supervisors to with senior executives. Specialty programs also exist for key positions such as operations shift managers and newly appointed directors on nuclear utility boards of directors.

In 2020, INPO developed virtual courses for topics such as coaching, decisionmaking and managing conflict. These programs have enabled INPO to influence industry leaders more frequently and in higher numbers. Although the programs were originally created in response to the pandemic, they have proved valuable and will be retained.

The National Academy for Nuclear Training e-Learning (NANTeL) system provides Web-based courses and proctored examinations across the industry. The NANTeL includes courses in areas such as unescorted plant access, radiation worker, industrial safety, maintenance, and engineering training and qualifications.

7.5 <u>Analysis and Information Exchange Programs</u>

Station organizations are required to share operating experience and lessons learned with INPO. The Institute then analyzes and communicates the information to the industry through various methods and products. In addition, INPO analyzes data to detect trends in industry performance and communicates the results to the industry.

The Institute 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 IRIS, a Web-based system, to exchange information on the safe operation of nuclear plants.

7.5.1 Operating Experience Program

The Institute 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 may be disseminated to the industry in IERs at one or more of the following levels:

- *Level 1 IER:* Provides recommendations for actions based on one or more significant industry events, an important industry issue, or an adverse trend. Level 1 IER recommendations constitute a new industry standard of performance.

- *Level 2 IER:* Highlights an area of concern based on an industry event or adverse trend that has broad applicability to several stations, which may or may not derive from significant events but has high consequence to plant safety or operation. Level 2 IER recommendations constitute a new industry standard of performance.

- *Level 3 IER:* Provides industrywide notification of important events and associated lessons. Level 3 IERs do not contain recommendations.

- *Level 4 IER:* Provides analysis of notable trends of equipment or human performance problems or other industrywide issues intended to heighten industry awareness. Level 4 IERs do not contain recommendations.

Members support the Operating Experience Program by providing INPO with detailed and timely operating experience information. Operating experience information is freely shared among INPO members through IRIS. IRIS entries enable a single station organization to multiply its experience base for identifying problems. This includes safety systems, which have similar components across many stations. A key to success is timeliness of reporting. INPO uses the following graded approach to prioritize event reports:

- *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 the discovery of the event or 10 days after the end of the month in which the event was discovered.

- *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 the recommendations provided in Level 1 and Level 2 IERs. During onsite plant evaluations, INPO follows up on the effectiveness of each station organization’s actions in response to these recommendations. Recent Level 1 and Level 2 IERs have covered topics such as operations and maintenance fundamentals and prevention of debris-induced fuel failures.

Members should review and take actions, as appropriate, on Level 3 and Level 4 IERs. The Institute evaluates each utility program’s effectiveness in extracting and applying lessons learned from industrywide and internal station operating experience.

The Institute 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. The Institute also provides just-in-time summaries in numerous topical areas, in a format designed to help plant personnel prepare for specific tasks. These documents provide ready-to-use materials to brief workers on problems experienced and lessons learned during recurring activities.

7.5.2 Development of Documents and Products

INPO creates many documents and other products to help member utilities and participants improve operations, maintenance, training, and other support for their nuclear plants. These documents are organized in the following categories:

- *Tier 1—Excellence Documents*: These establish the standards that INPO members and participants are expected to meet, and by which INPO evaluates station performance. The following are examples of Tier 1 documents:

  <u>POs&Cs</u>, which are common for INPO and WANO, are the functional area standards of excellence for stations and corporate organizations. These describe excellence in the operations, maintenance, support, and governance of commercial nuclear power plants. The “objective” portion is the standard, while the associated “criteria” further describe each objective. The POs&Cs support the attainment of the following operational excellence outcomes:

  - sustainable, high-level plant performance
  - sustainable, event-free operation
  - avoidance of unplanned, long-duration shutdowns
  - well-understood and well-managed safety, design, and operational margins
  - high levels of worker safety
  - a highly skilled, knowledgeable, and collaborative workforce

  <u>Principles Documents</u> describe the attributes, traits, and behaviors associated with important industry themes and issues. Recent examples of Principles documents that established new standards of excellence include INPO 15-005, “Leadership and Team Effectiveness Attributes,” issued September 2016; INPO 12-008, “Principles for Excellence in Integrated Risk Management,” Revision 1, issued August 2013; INPO 17-004, “Principles for Excellence in Corporate Performance,” issued October 2017, and INPO 19-003. In some instances, Principles documents are developed to augment objectives and criteria. Principles documents may eventually be incorporated into subsequent revisions of POs&Cs, or they may stand alone for a long period.

  <u>Level 1 and Level 2 IERs</u> are covered in the section “Operating Experience Program” above, which gives detailed expectations for their review and use.

- *Tier 2 — Supporting and Implementing Documents*: These provide information intended 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. The following are examples of Tier 2 documents:

  <u>Level 3 IERs</u> are covered in the section “Operating Experience Program” above, which gives detailed expectations for their review and use.

  <u>Guidelines</u> provide specific information and activities important to achieving standards of

excellence as outlined in the related objectives and criteria. The documents provide additional guidance and detail considered necessary to fully 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 through defined processes. The “AP” annotation originally stood for “advanced plant”; however, the reference has gradually changed to “advanced process.” These are evolutionary documents that incorporate current industry best practices.

Operating Experience Program descriptions provide an overview of the INPO-sponsored Operating Experience Program and its expectations for INPO and INPO members.

- *Tier 3—Other Documents*: These are documents that are not in Tier 1 or Tier 2 of this hierarchy. Information in this tier provides reference or amplifying information on various topics for review and discretionary use by INPO members and participants. The information may be created by an organization other than INPO. Tier 3 information varies greatly in format and style and may not be subjected to the strict document production quality controls required for Tier 1 and Tier 2 documents. The following are examples of Tier 3 documents:

  Level 4 IERs are covered in the section “Operating Experience Program” above, which gives detailed expectations for their review and use.

  Good Practices provide examples of effective methods for accomplishing elements of nuclear plant management and operation.

  Manuals are collections of data or other information for use by INPO members and participants. The documents provide a convenient collection of concepts, insights, and suggested activities to assist station personnel in understanding, implementing and performing particular station functions.

  Reports provide descriptions and results of INPO or INPO-sponsored activities of broad interest to the industry, including 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

The Institute 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.

7.5.3 Workshops and Meetings

The Institute sponsors workshops and working meetings for specific groups of managers on specific technical issues. These activities serve as forums for information exchange, allowing INPO and industry personnel to discuss challenges, performance issues, and areas of interest. They also allow INPO members and participants to meet and exchange information with their counterparts. In 2021, more than 4,000 industry personnel participated in more than 105 seminars, workshops, and technical working meetings at INPO.

7.5.4 Data Collection and Trending

The Institute operates and maintains IRIS as the single repository for data and information related to nuclear plant performance. Members provide routine operational data monthly, in accordance with the INPO, WANO, and NRC performance indicator programs or regulatory requirements. Plant data then undergoes analysis and trending. Members access this data to compare their performance with other nuclear stations, to track progress toward specific performance goals, and to monitor the performance of their nuclear plants.

Over the years, specific indicators have been developed that have safety and reliability significance and that are utilized industrywide for performance tracking and comparison. Examples include number of unplanned automatic scrams, safety system performance, unit capability factor, number of forced losses of generation, fuel reliability, collective radiation exposure levels, and industrial safety accident rates. Beginning in 1990, the industry established goals for each of these important indicators. The indicators and goals are reviewed and updated every 5 years.

7.5.5 Equipment Performance Data

The industry reports equipment performance data to IRIS. Member utility organizations use the data to identify and address performance problems, with the goal of continuously improving plant safety and reliability. The Institute also uses the equipment data for performance trending, to identify industrywide equipment challenges that need to be addressed. The Institute also makes this data available to the NRC, to support the NRC's equipment performance reviews.

7.5.6 Operating Experience for New Plant Construction

In 2009, INPO established a means for collecting and distributing experience from plants under construction 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 new construction projects.

7.5.7 Neural Plant Performance Indicator

In 2014, INPO began to utilize Plant Performance Indicator (PPI), which is a statistical model that provides a numerical value for a station's current performance, correlated to an INPO assessment score. The PPI is updated quarterly and shared with the industry in plant performance summaries. In subsequent upgrades, the PPI has incorporated neural modeling techniques, which has improved its accuracy and responsiveness. Individual functional area models have also been developed, which provides a more detailed view of current performance.

The Neural PPI has been useful in tracking a station's current performance in between INPO evaluations and WANO peer reviews. This has made it a valuable addition to the all-source information that is utilized in continuous performance monitoring.

7.5.8 Other Analysis Activities

The Institute analyzes industry operational data from various sources—events, equipment failures, performance indicators, and regulatory reports—to detect trends in industry performance. The Institute communicates the results of analyses and suggested actions to the industry. Subjects of recent analyses include common contributors to equipment problems, an adverse trend in primary pump seal failures, an adverse trend in debris-related nuclear fuel failures, and weaknesses in handling highly radioactive filters. Station organizations use this information to assess their performance and to identify improvement opportunities. Finally, plant performance data are analyzed, and the results are used to support other INPO operations, such as evaluations and assistance activities.

7.6 Member Support Missions

Station organizations can request and receive assistance in specific problem areas to help improve performance. Resources are provided using a graded approach, with higher priority going to lower performing plants. This support is targeted for specific technical challenges and for broader management and organizational drivers that may underlie gaps in performance. While organizations usually initiate assistance requests, INPO may also suggest support missions in specific areas requiring improvement.

Personnel from INPO 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 specific maintenance task particularly well.

The Institute provides the requesting utility with written reports that detail the results of the visit. In most cases, member support missions include the provision of plans and methods for improving performance and are not purely evaluative in nature. Effectiveness reviews reveal that the visits are highly valued by station leaders and that they contribute to improved performance.

7.7 Special Focus Program

There is a direct correlation between station performance and the probability of an adverse event, such that very-low-performing stations typically experience consequential or even significant operational events. To prioritize assistance to these stations, in 2005 INPO created the Special Focus Program.

Since the inception of the program, INPO has improved the methods, tools, and training used to sustainably recover performance at low-performing stations. Methods include structured interactions with station leaders, utility executives, CEOs, and even the boards of directors. At INPO, specific individuals are earmarked to participate on recovery teams based on their broad experience, leadership attributes, and ability to communicate effectively up and down the chain of command. Formal assistance plans may call for specialized assistance teams comprised of experts from both INPO and the industry; such plans are executed and monitored under the leadership of an INPO Director of Recovery.

As stated earlier, the Institute has revised Policy Note 14 and has officially implemented its continuous monitoring engagement categories—Monitoring, Augmented Monitoring, and Full Monitoring (with Special Focus is a subset of Full Monitoring)—for U.S. stations and corporations. The change was made to highlight the role of continuous monitoring in developing a deeper, more comprehensive picture of performance and sustainability, allowing INPO to better focus on stations and corporations at greatest risk of decline.

Historically, it was commonplace for 10 or 12 out of a total of about 60 stations to be categorized as low performers and Special Focus plants. Some stations would improve performance and be removed from the Special Focus Program only to re-enter as performance subsequently degraded. Other stations would linger at low levels and remain in the program for several years. As Special Focus methods matured, INPO's ability to recover outliers improved, and the number of Special Focus stations decreased. As of January 2022, there were no U.S. stations and only one non-U.S. WANO-AC station in Special Focus.

The Institute is now applying a formal innovation process to identify enhancements for recovering station performance faster and more sustainably. Specifically, it is addressing its ability to influence the performance of State-governed utilities that are heavily influenced by local cultures.

8 Relationship with the World Association of Nuclear Operators

The Institute represents U.S. nuclear utilities as a Category 1 member of WANO. The Institute also operates WANO-AC, one of four global WANO regional centers. As part of the international strategy, INPO influences, shares, and supports WANO's strategic goals (including the Action for Excellence initiative) and coordinates many U.S. nuclear utility activities with WANO.

As part of the operating model, INPO programs fully meet the requirements of WANO programs and processes. 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 United States based on lessons learned from the more than 355 units that are operated outside of the United States. The Institute works to remain fully aware of trends in the global nuclear industry and continues to strengthen relationships in this area.

Key programs and activities for INPO/WANO-AC include the following:

- As a WANO member, INPO performs station and corporate peer reviews, member support missions, and other WANO activities worldwide and invites international participants from other WANO members to take part in INPO activities.

- The Institute and WANO-AC is a key source of support, best practices, and operating experience for WANO as it transitions its programs and operations to help members worldwide achieve the 2030 goals of the Action for Excellence initiative.

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

- The Institute interacts with and facilitates improvement of similar organizations, including other national-level self-regulators such as the Japan Nuclear Safety Institute and the China Nuclear Energy Association. Additionally, INPO/WANO-AC works with the International Atomic Energy Agency, the Nuclear Energy Agency , and the other WANO regional centers so that synergies in operational safety approaches are realized.

- As part of the WANO New Unit Assistance program, INPO/WANO-AC provides services and products to support safe and reliable startup of new WANO-AC 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. Since 2019, WANO-AC members in the United Arab Emirates (Barakah Nuclear Energy Plant, Units 1 and 2) and China (Haiyang Nuclear Power Plant and Shidao Bay Nuclear Power Plant) have successfully transitioned units from construction to the operational phase. Additional units in the United States (Vogtle Electric Generating Plant, Units 3 and 4), United Arab Emirates (Barakah Units 3 and 4), and China (State Nuclear Power Demonstration Plant, Units 1 and 2) are scheduled to commence operations in the next few years. Once a unit becomes operational, it enters the Continuous Performance Monitoring Program, described earlier in this document. The INPO/WANO-AC New Unit Assistance program is scalable and adaptable to meet the anticipated needs of potential small modular reactor members.

WANO-AC members include the following:

- Bruce Power (Canada)
- Centrala Nuclearelectrica (Romania)
- China Huaneng Group (China)
- Comisión Federal de Electricidad (Mexico)
- Emirates Nuclear Energy Corporation (United Arab Emirates)
- Eskom Holdings (South Africa)
- New Brunswick Power (Canada)
- Ontario Power Generation (Canada)
- State Power Investment Corporation (China)
- All U.S. nuclear utilities

9 Industry Response to the Accident at Fukushima

A coordinated effort of EPRI, INPO, and the NEI—in conjunction with senior utility executives—created a joint leadership model to respond to events at the Fukushima Daiichi nuclear energy facility. This model ensured 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's response has been to maintain and improve its already high levels of operational safety and reliability, while applying the lessons from the Fukushima Daiichi 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. The Institute has verified that actions have been completed through various review activities, including INPO evaluations and WANO peer reviews, emergency management performance evaluations, and in-office document reviews. The sustainability of the actions is reviewed during WANO peer reviews.

The Institute developed training materials to assist utilities in preparing for beyond-design-basis events. These materials include case studies and instructor-led training focused on decisionmaking and decisionmaking under stress. A guideline for establishing effective training for emergency response personnel was also developed. The guideline includes the results of a job analysis that identified the 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.

INPO has upgraded its emergency plan and emergency response facilities to help members mobilize industry resources to assist a site experiencing an event. The INPO member utilities signed a mutual assistance agreement to provide resources during events if requested. The Institute conducts quarterly drills, most involving the NEI and EPRI, to practice response actions. Some are conducted in conjunction with utilities' during their regularly scheduled emergency preparedness drills.

9.1 Diverse and Flexible Coping Strategies

In response to the Fukushima accident, the Institute and the NEI 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 that result in an extended loss of AC power and a loss of normal access to the ultimate heat sink.

The Institute and its members established an indefinite coping capability by relying on installed equipment, on-site portable equipment, and prestaged offsite resources. The equipment ranges from diesel-driven pumps and electric generators to ventilation fans, hoses, fittings, cables, and communications gear. Equipment is stored at strategic locations and protected to ensure it can be used if other systems involved in 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 assumes that onsite resources must be sufficient to cope for the first 24 hours after an event. A standardized list of equipment connectors was developed to address interchangeability of equipment. Each site is required to have one set of FLEX equipment on site for each unit, plus one extra set. This makes each site a potential source of FLEX equipment for other sites. During an emergency, a call to INPO or directly to another station will activate mobilization of FLEX equipment.

In addition, there are two large response centers that can deliver equipment to any U.S. site within a defined length of time. The response centers are managed by a vendor, the Strategic Alliance for FLEX Emergency Response, which was created jointly by the Pooled Equipment Inventory Company and AREVA. Each response center has five sets of FLEX equipment: four sets to support sites and one set that may be 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 four decades since the Three Mile Island accident. The leaders who guided the industry over these decades of challenge and change showed great insight in recognizing the need for an unprecedented form of industry self-regulation, as established 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's commitment to improved performance has provided the foundation for a unique, sustained partnership between INPO and its members. The Institute 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. The Institute does not take credit for this success, but it does take pride in its contributions to the industry it serves.

The Institute 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 competitive, complex, and increasingly global business environment.

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 the utmost integrity and an unrelenting drive toward excellent performance.

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APPENDIX B - U.S. COMMERCIAL NUCLEAR POWER REACTORS

SOURCE: U.S. Nuclear Regulatory Commission, NUREG-1350, “2021–2022 Information Digest,” Volume 33, August 2021.

NOTE: Since the issuance of the 2019 U.S. National Report, five units have ceased operations (i.e., Indian Point Nuclear Generating, Units 2 and 3; Duane Arnold Energy Center; Three Mile Island Nuclear Station, Unit 1; and Palisades), bringing the total to 92 operating commercial nuclear facilities in the United States.

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Arkansas Nuclear One, Unit 1—Entergy Operations, Inc.	PWR	2,568	12/74–05/34
Arkansas Nuclear One, Unit 2—Entergy Operations, Inc.	PWR	3,026	03/80–07/38
Beaver Valley Power Station, Unit 1—Energy Harbor Nuclear Generation LLC/Energy Harbor Nuclear Corporation	PWR	2,900	10/76–01/36
Beaver Valley Power Station, Unit 2—Energy Harbor Nuclear Generation LLC/Energy Harbor Nuclear Corporation	PWR	2,900	11/87–05/47
Braidwood Station, Unit 1—Exelon Generation Company, LLC	PWR	3,645	07/88–10/46
Braidwood Station, Unit 2—Exelon Generation Company, LLC	PWR	3,645	10/88–12/47
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—Duke Energy Progress, LLC	BWR	2,923	03/77–09/36
Brunswick Steam Electric Plant, Unit 2—Duke Energy Progress, LLC	BWR	2,923	11/75–12/34
Byron Station, Unit 1—Exelon Generation Company, LLC	PWR	3,645	09/85–10/44
Byron Station, Unit 2—Exelon Generation Company, LLC	PWR	3,645	08/87–11/46
Callaway Plant, Unit 1—Union Electric Company	PWR	3,565	12/84–10/44

Plant Name and Licensee	**Reactor Design Type**	**Licensed Power (MWt)**	**Operating Lifetime**
Calvert Cliffs Nuclear Power Plant, Unit 1—Calvert Cliffs Nuclear Power Plant, LLC/Exelon Generation Company, LLC	PWR	2,737	05/75–07/34
Calvert Cliffs Nuclear Power Plant, Unit 2—Calvert Cliffs Nuclear Power Plant, LLC/Exelon Generation Company, LLC	PWR	2,737	04/77–08/36
Catawba Nuclear Station, Unit 1—Duke Energy Carolinas, LLC	PWR	3,469	06/85–12/43
Catawba Nuclear Station, Unit 2—Duke Energy Carolinas, LLC	PWR	3,411	08/86–12/43
Clinton Power Station, Unit 1—Exelon Generation Company, 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—Comanche Peak Power Company LLC/Vistra Operations Company LLC	PWR	3,612	08/90–02/30
Comanche Peak Nuclear Power Plant, Unit 2—Comanche Peak Power Company LLC/Vistra Operations Company LLC	PWR	3,612	08/93–02/33
Cooper Nuclear Station—Nebraska Public Power District	BWR	2,419	07/74–01/34
Davis-Besse Nuclear Power Station, Unit 1—Energy Harbor Nuclear Generation LLC/Energy Harbor Nuclear Corporation	PWR	2,817	07/78–04/37
Diablo Canyon Power Plant, Unit 1—Pacific Gas & Electric Company	PWR	3,411	05/85–11/24
Diablo Canyon Power Plant, Unit 2—Pacific Gas & Electric Company	PWR	3,411	03/86–08/25
Donald C. Cook Nuclear Plant, Unit 1—Indiana Michigan Power Company	PWR	3,304	08/75–10/34
Donald C. Cook Nuclear Plant, Unit 2—Indiana Michigan Power Company	PWR	3,468	07/78–12/37
Dresden Nuclear Power Station, Unit 2—Exelon Generation Company, LLC	BWR	2,957	06/70–12/29
Dresden Nuclear Power Station, Unit 3—Exelon Generation Company, LLC	BWR	2,957	11/71–01/31
Edwin I. Hatch Nuclear Plant, Unit 1—Southern Nuclear Operating Company Inc.	BWR	2,804	12/75–08/34
Edwin I. Hatch Nuclear Plant Unit 2—Southern Nuclear Operating Company, Inc.	BWR	2,804	09/79–06/38

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
Fermi, Unit 2—DTE Electric Company	BWR	3,486	01/88–03/45
R.E. Ginna Nuclear Power Plant—R.E. Ginna Nuclear Power Plant, LLC	PWR	1,775	07/70–09/29
Grand Gulf Nuclear Station, Unit 1—Entergy Operations, Inc.	BWR	4,408	07/85–11/44
H.B. Robinson Steam Electric Plant, Unit 2—Duke Energy Progress, LLC	PWR	2,339	03/71–07/30
Hope Creek Generating Station, Unit 1—PSEG Nuclear, LLC	BWR	3,902	12/86–04/46
James A. FitzPatrick Nuclear Power Plant—Exelon Generation Company, LLC	BWR	2,536	07/75–10/34
Joseph M. Farley Nuclear Plant, Unit 1—Southern Nuclear Operating Company, Inc.	PWR	2,775	12/77–06/37
Joseph M. Farley Nuclear Plant, Unit 2—Southern Nuclear Operating Company, Inc.	PWR	2,775	07/81–03/41
La Salle County Station, Unit 1—Exelon Generation Company, LLC	BWR	3,546	01/84–04/42
La Salle County Station, Unit 2—Exelon Generation Company, LLC	BWR	3,546	10/84–12/43
Limerick Generating Station, Unit 1—Exelon Generation Company, LLC	BWR	3,515	02/86–10/44
Limerick Generating Station, Unit 2—Exelon Generation Company, LLC	BWR	3,515	01/90–06/49
McGuire Nuclear Station, Unit 1—Duke Energy Carolinas, LLC	PWR	3,411	12/81–06/41
McGuire Nuclear Station, Unit 2—Duke Energy Carolinas, LLC	PWR	3,411	03/84–03/43
Millstone Power Station, Unit 2—Dominion Energy Nuclear Connecticut, Inc.	PWR	2,700	12/75–07/35
Millstone Power Station, Unit 3—Dominion Energy Nuclear Connecticut, Inc.	PWR	3,650	04/86–11/45
Monticello Nuclear Generating Plant, Unit 1—Northern States Power Company—Minnesota	BWR	2,004	06/71–09/30
Nine Mile Point Nuclear Station, Unit 1—Nine Mile Point Nuclear Station, LLC/Exelon Generation Company, LLC	BWR	1,850	12/69–08/29
Nine Mile Point Nuclear Station, Unit 2—Nine Mile Point Nuclear Station, LLC/Exelon Generation Company, LLC	BWR	3,988	03/88–10/46
North Anna Power Station, Unit 1—Virginia Electric & Power Company	PWR	2,940	06/78–04/38

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
North Anna Power Station, Unit 2—Virginia Electric & Power Company	PWR	2,940	12/80–08/40
Oconee Nuclear Station, Unit 1—Duke Energy Carolinas, LLC	PWR	2,610	07/73–02/33
Oconee Nuclear Station, Unit 2—Duke Energy Carolinas, LLC	PWR	2,610	09/74–10/33
Oconee Nuclear Station, Unit 3—Duke Energy Carolinas, LLC	PWR	2,610	12/74–07/34
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 Generation Company, LLC	BWR	4,016	07/74–08/53
Peach Bottom Atomic Power Station, Unit 3—Exelon Generation Company, LLC	BWR	4,016	12/74–07/54
Perry Nuclear Power Plant, Unit 1—Energy Harbor Nuclear Generation LLC/Energy Harbor Nuclear Corporation	BWR	3,758	11/87–03/26
Point Beach Nuclear Plant, Unit 1—NextEra Energy Point Beach, LLC	PWR	1,800	12/70–10/30
Point Beach Nuclear Plant, Unit 2—NextEra Energy Point Beach, LLC	PWR	1,800	10/72–03/33
Prairie Island Nuclear Generating Plant, Unit 1—Northern States Power Company—Minnesota	PWR	1,677	12/73–08/33
Prairie Island Nuclear Generating Plant, Unit 2—Northern States Power Company—Minnesota	PWR	1,677	12/74–10/34
Quad Cities Nuclear Power Station, Unit 1—Exelon Generation Company, LLC	BWR	2,957	02/73–12/32
Quad Cities Nuclear Power Station, Unit 2—Exelon Generation Company, LLC	BWR	2,957	03/73–12/32
River Bend Station, Unit 1—Entergy Operations, Inc.	BWR	3,091	06/86–08/45
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—NextEra Energy Seabrook, LLC	PWR	3,648	08/90–03/50

Plant Name and Licensee	Reactor Design Type	Licensed Power (MWt)	Operating Lifetime
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—Duke Energy Progress, LLC	PWR	2,948	05/87–10/46
South Texas Project, Unit 1—STP Nuclear Operating Company	PWR	3,853	08/88–08/47
South Texas Project Unit 2—STP Nuclear Operating Company	PWR	3,853	06/89–12/48
St. Lucie Plant, Unit 1—Florida Power & Light Company	PWR	3,020	12/76–03/36
St. Lucie Plant, Unit 2—Florida Power & Light Company	PWR	3,020	08/83–04/43
Surry Power Station, Unit 1—Virginia Electric & Power Company	PWR	2,587	12/72–05/52
Surry Power Station, Unit 2—Virginia Electric & Power Company	PWR	2,587	05/73–01/53
Susquehanna Steam Electric Station, Unit 1—Susquehanna Nuclear, LLC	BWR	3,952	06/83–07/42
Susquehanna Steam Electric Station, Unit 2—Susquehanna Nuclear, LLC	BWR	3,952	02/85–03/44
Turkey Point Nuclear Generating, Unit 3—Florida Power & Light Company	PWR	2,644	12/72–07/52
Turkey Point Nuclear Generating, Unit 4—Florida Power & Light Company	PWR	2,644	09/73–04/53
Virgil C. Summer Nuclear Station—Dominion Energy South Carolina Inc.	PWR	2,900	01/84–08/42
Vogtle Electric Generating Plant, Unit 1—Southern Nuclear Operating Company, Inc.	PWR	3,625	06/87–01/47
Vogtle Electric Generating Plant, Unit 2—Southern Nuclear Operating Company Inc.	PWR	3,625	05/89–02/49
Waterford Steam Electric Station, Unit 3—Entergy Operations, Inc.	PWR	3,716	09/85–12/44
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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10. SUPPLEMENTARY NOTES

This report is an update to NUREG-1650, Revision 7

11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission has prepared Revision 8 to NUREG-1650, “The United States of America Ninth National Report for the Convention on Nuclear Safety,” for submission for peer review at the joint eighth and ninth review meeting of the Convention on Nuclear Safety, to be convened at the International Atomic Energy Agency in Vienna, Austria, in March 2023. 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, the 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 2019, 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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